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ENVIRONMENTAL IMPACT OF NUCLEAR INSTALLATIONS

PROCEEDINGS OF THE JOINT SEMINARY
FROM SEPTEMBER 15th TO 18th 1992
AT THE UNIVERSITY OF FRIBOURG/SWITZERLAND

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Organised by
the

Société Française de Radioprotection

and the

German-Swiss Fachverband für Strahlenschutz

Hansruedi Völkle and Serge Prêtre, editors

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Zone Industrielle de Courtabœuf,
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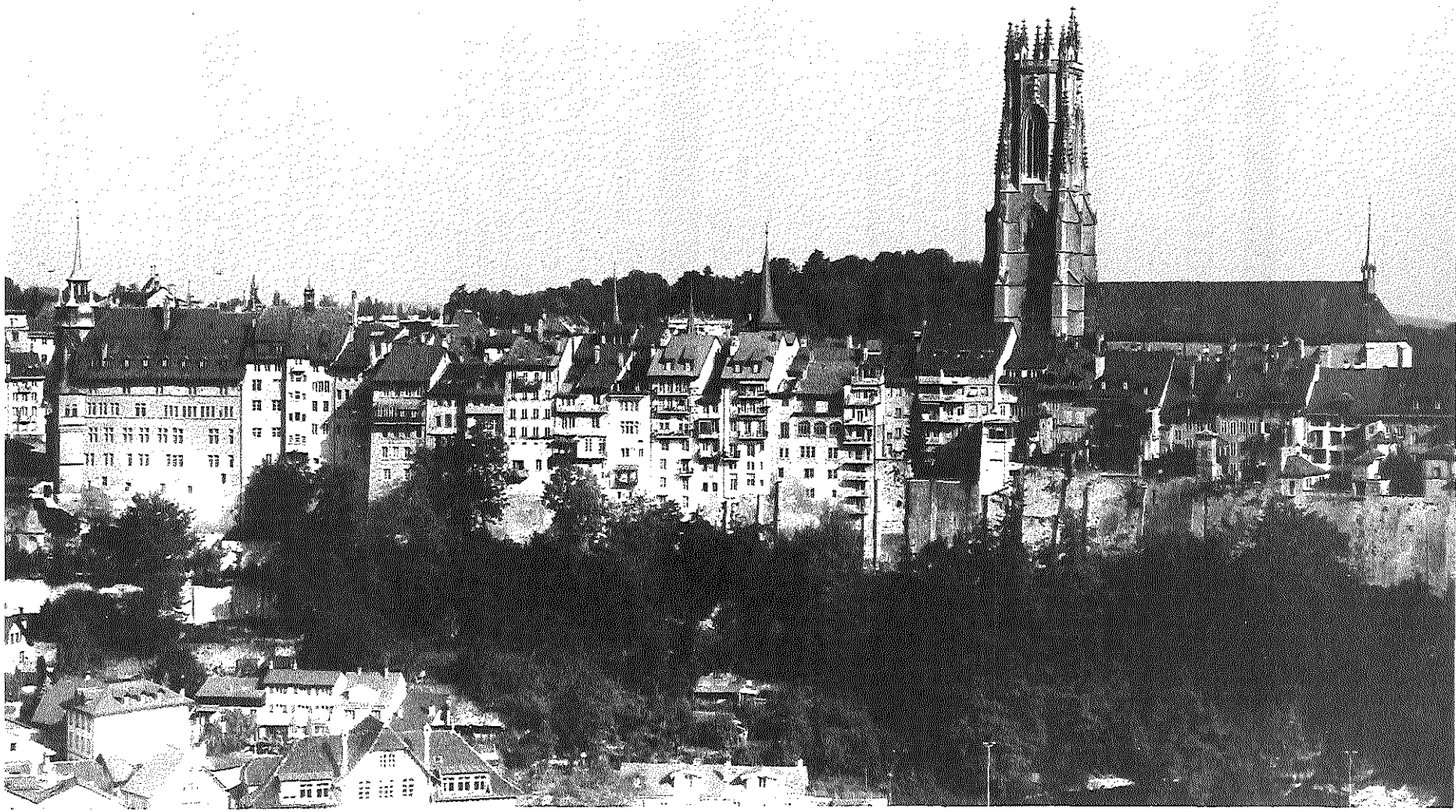

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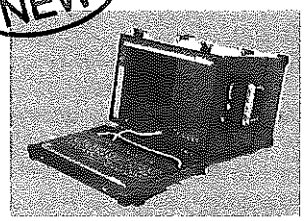

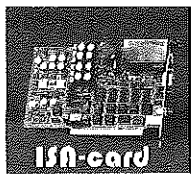

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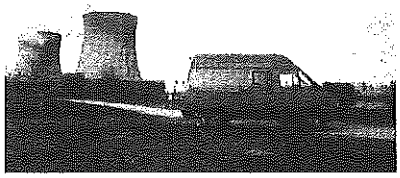

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Preface

The medieval city of Fribourg was founded in the 12th century in the western part of Switzerland by the Duke of Zähringen and is situated on limestone rock between the innumerable slopes of the Saane river. This naturally ideal place for a fortified city of the middle ages needed, in the 19th and 20th centuries - when trade, industry, traffic and communication became important - the construction of not less than fourteen bridges to open the town to the surrounding countryside. As the river Saane marks also the border between the German- and the French-speaking parts of the country, the bilingual city of Fribourg/Freiburg became, in the figurative sense also, a "bridge-town" between two cultures and languages. Fribourg University, founded in 1891 by the statesman Georges Python, with today some 7000 students from over 90 nations, has from the beginning always paid great attention to international scientific contacts, especially when calling lecturers and students from abroad.

Political changes in eastern European countries and the unification process within the growing European Community are stimulating Switzerland to play a more active role in Europe. Our country should therefore intensify international scientific exchange and co-operation in the field of radiation protection, particularly concerning radiation protection norms, monitoring programmes, measurement techniques and protective measures for nuclear accidents.

Nuclear electricity plays an important role in the energy production in many countries, in some more than 50% of the total electricity produced. This role could became even more important in the future, if we have to substitute fossil energy (fuel, coal and gas) for environmental protection reasons and reduce carbon dioxide releases to the atmosphere. On the other hand, from the Chernobyl accident we learned that every technical installation has its risks and dangers, which have to be considered in comparison with other risks of our modern civilisation. An optimisation process is therefore necessary between risks and benefits of every new technology. Technical installations have to be constantly controlled and improved in relation to internal security, protection of the workers and environmental impact.

These few points were, among others, the reasons for this **Joint Scientific Meeting** on the **Environmental Impact of Nuclear Installations,** organised by the French and the German-Swiss Radiation Protection Societies, the ***Société Française de Radioprotection*** and the ***Fachverband für Strahlenschutz*** at Fribourg University from 15th to 18th September 1992. The conference was attended by about 300 radiation protection specialists, mainly from Germany, France and Switzerland. Local organiser of the symposium was the ***Branch "Surveillance of Radioactivity"*** of the ***Federal Office of Public Health***, whose laboratories have been located at the Physics Department of the University of Fribourg for more than thirty years. Parallel to the conference, twenty companies and institutions presented their latest radiation protection measuring instruments, monitoring systems for environmental radioactivity, computer programmes for data processing, dose rate calculations and emergency pre-

cautions in a well attended exhibition which was held in the lobby of the institute.

The aim of the conference was to present the state of the art in monitoring of nuclear installations and the assessment of their radiological impact on environment and population; technical and organisational preparative provisions for accidents in nuclear power plants and possibilities of dealing with the consequences of a nuclear accident were discussed as well. The conference closed with a panel discussion which focused, in particular, on the co-operation across borders in the event of accidents in nuclear power plants and the role of the media in informing the public on radiation risks and radiation protection. One essential goal of the conference was the intensification of the co-operation in radiation protection between central European countries. More specific goals were, on one hand, the rapid international exchange of data and experiences, and, on the other hand, a further harmonisation of the measurement methods and the monitoring networks, aiming at assessing, limiting and reducing radiation doses in case of emergencies and accidents.

The conference focused on four issues, the first being the **monitoring and balancing of the radioactivity discharges** from nuclear installations into the environment. Germany, France and Switzerland reported unanimously that in recent years the radiation doses for the personnel in nuclear power plants as well as the radioactivity discharges into the environment, and thus the radiation exposure of the population, were further reduced, due to continuous technical innovations and improvements of the facilities, and due to the high level of training of the personal. In most cases, therefore, artificial environmental radioactivity cannot be detected by the most sensitive measuring methods. The radiation doses of the population living in the vicinity of nuclear facilities lie, even under the most adverse assumptions, no higher than one to two percent of the natural radiation doses (without Radon).

The second topic was concerned with model calculations, simulation methods and computer programmes for the **dispersion of radioactivity in the environment** and the resulting radiation doses. These are important, on the one hand, in order to prepare prognoses for the environmental impacts and radiation doses of the population caused by the radioactivity discharges during normal operation, and, on the other hand, to simulate model scenarios for nuclear power plant accidents in order to optimise the protective measures for the public. Improved models, wind channel experiments and powerful computers allow more accurate predictions today, in the immediate vicinity as well as over large distances.

The third topic focused on **monitoring methods and measurement programmes** of environmental radioactivity. In addition to sampling in the field with subsequent measurement of the samples in the laboratory, the direct measurement at site, the so-called in-situ-measurements, is gaining in importance. Furthermore, thanks to the use of modern computer and remote transfer technique, one can now monitor certain variables directly at the site by

means of automatic instruments, and the results can be transferred continuously to a central data station. Such automatic networks for monitoring the ambient dose or the particulate radioactivity in air, respectively, are being set up in several European countries, or are already in operation. Their data can directly be consulted by the public (by TELETEXT in Switzerland or MINITEL in France). Some papers were concerned with the dispersion of radioactivity in aqueous systems, with a particular emphasis on measuring programmes and results in the rivers Rhine, Rhone, Danube and Tagus (Portugal), as well as in the sea.

The fourth topic was concerned with preparative measures and dealing with the consequences of **emergencies and accidents** in nuclear installations. Emergency organisation, operation plans and protective measures of all three countries were presented; a very important role plays the rapid and non-bureaucratic, bilateral co-operation with regard to installations close to the border. Computer programmes were presented which help the authorities to select the most suitable protective measure for the momentary situation (radiation contamination, harvest situation, supplies, distribution possibilities etc.) as fast as possible. The option of a filtered and controlled pressure release will help to significantly reduce the radiological impact on environment and population in case of certain serious nuclear power plant accidents. The operation of German, French and Swiss measuring teams in Russia and Ukraine, as well as a French-Russian co-operation on experiments for ground decontamination were also reported. For example, a kind of lawn is planted which later will be rolled up like a carpet, thus removing a few centimetres of top soil and eliminating a substantial portion of the ground contamination.

The importance of **co-operation across borders** in achieving harmonisation of monitoring methods and for the assessment of radiation doses and measurements was again emphasised in the concluding **panel discussion.** In addition to quick informal contacts between state and regional authorities, on the one hand, and emergency organisations, on the other hand, the direct access to the databases of the monitoring networks of neighbouring countries is of great importance. Across-the-border practices including observers and experts from neighbouring countries are a step toward fulfilling these need. Many radiation protection professionals obviously are not very happy with the role of the media in **informing the public** on radioactivity, radiation and their risks. One reason is certainly that the experts have the information but are not able to give it to the journalists in a manner that is comprehensible and suitable for the public; on the other hand, some journalists are more interested in, from their point of view, certain sensational aspects of a particular information rather than in the actual information itself. This kind of "communication" problems can be avoided only by more personal contracts and more intensive discussion of the journalists directly with the experts in order to provide the public with factual information on radioactivity, radiation and their risks. ***Radiation protection professionals should therefore intensify their contact with the media and the public, should clearly point out the risks and dangers wherever there are any, set these risks***

in relation to other risks of everyday life, and oppose any tendency of scaremongering where there are no risks at all.

We thank all participants for attending the meeting and the authors for their scientific contributions. The 94 manuscripts, prepared ready for printing by the authors, are published in this extra volume of "RADIOPROTECTION" the official journal of the SFRP.

Fribourg and Villigen in November 1992

Hansruedi VÖLKLE
Congress Secretary

Serge PRETRE
Congress President

Part 1: Radioactivity Release Monitoring

MANAGEMENT OF RADIOACTIVE WASTE RELEASES FROM *EDF* PWR NUCLEAR POWER PLANTS

Mme Rufenach, M. P. Hartmann, M. T. Fugain
Electricité de France, 93203 Saint Denis, Cedex 1

SUMMARY

Since 1980, EDF has undertaken a number of actions to reduce releases of radioactive wastes from nuclear power plants to a level as low as reasonably achievable (the ALARA principle). These actions (improvement of the collecting and treatment circuits as well as implementation of a thorough management system) have made it possible to reduce by a significant factor the volumes of liquid waste arising, and as a consequence, the releases of radioactivity excluding tritium by a factor of 10 in ten years.

These good results are also accompanied by a palpable reduction in the volume of solid releases linked to treatment and, therefore, of the corresponding costs.

1 - INTRODUCTION

Like many industrial activities, the operation of nuclear power plants entails the production of effluents which are released into the environment. Among these, the liquid and gaseous radioactive effluents are subject to a rigorous monitoring procedure.

Whether they originate directly from reactors or from the circuits of nuclear auxiliaries, these effluents are collected, processed and analysed prior to their release into the environment in accordance with the strict terms and conditions laid down by French regulations.

These regulations, comprising texts of a general nature: decrees of 6th November 1974 for gaseous wastes and of 31st December, 1974, for liquid wastes, are supplemented by decrees specific to each site.

This document gives an assessment and the regulations for managing these releases of liquid and gaseous radioactive wastes from PWR nuclear power plants, which supply more than 75% of all electricity produced in France.

The set of PWR nuclear power plants operated by EDF now comprises fifty two reactors (thirty four 900 MW reactors and eighteen 1300 MW reactors) located at eighteen different sites.

2 - ORIGIN OF RADIOACTIVE EFFLUENTS PRODUCED [FIG. 1]

2.1 - Gaseous Effluents

Their Source and Nature

Gaseous effluents can be divided into two broad categories:

- hydrogenated gaseous effluents resulting from the degasing of primary wastes and the sweeping of tanks with gas blankets containing primary water which has not been degased. These effluents contain:
  - hydrogen and nitrogen,
  - gaseous fission products (krypton, xenon, iodine etc.).
- Aerated gaseous products arising from the collection of gases vented from the treatment circuits and from the tanks which take the spent liquid effluents. This collection process is performed by the ventilation circuits. Ventilation return air from the rooms of the nuclear island comprises, in terms of volume, the majority of the gaseous releases.

Treatment

The hydrogenated gaseous effluents are held in the tanks to allow their radioactivity to decay and to enable monitoring prior to release. The majority of gaseous fission products and any iodine have a radioactive half-life of less than one week: a regulatory storage period of one month makes it possible to reduce their radioactivity by at least a factor of ten. Release takes place via the stack after passage through filtration devices comprising absolute filters for the retention of aerosols and dust, and iodine traps for the halogen gases, if required.

Aerated gaseous effluents are released via the stack after filtration and dilution in air from the ventilation system.

Release into the Atmosphere

The gaseous releases and the ventilation air are released via the stack in accordance with conditions laid down by French regulations, which specifically stipulate:

- the release conditions:
  - prior analysis of the effluents in the hold-up tanks,
  - on-going monitoring of radiation being released via the stack using an alarmed measuring device which has a 4 MBq/m^3 threshold,
  - weekly samples from the devices making it possible to measure releases of halogens and aerosols,
- the superinduced atmospheric conditions which must not be exceeded in the natural environment (for gases, halogens and aerosols),

- annual radioactivity limits which must not be exceeded, in the case of gaseous radioactive elements, those for the halogens and aerosols,
- the nature of the monitoring procedures to be carried out in the environment:
  - monitoring of ambient radiation and of atmospheric dust,
  - monitoring of rainwater and plants.

2.2 - Liquid Effluents

Their Source and Nature

Liquid effluents can be divided into two broad categories:

- hydrogenated wastes, arising from the primary circuit. These result from movements of water initiated in the primary circuit, particularly during variations of the load and to compensate burn-up of the fuel. These effluents contain gases in solution (xenon, iodine etc.), fission products (cesium, iodine etc.), activation products (cobalt, manganese etc.) as well as chemical substances such as boric acid and lithium. These wastes can be recycled, if required.
- spent and non-recyclable effluents, among which figure:
  - activated and chemically clean effluents: these are the process tailings. They arise essentially from leaks of primary water, and from draining equipment,
  - radioactive and chemically polluted effluents: these are the chemical effluents resulting mainly from fuel removal operations and from decontamination processes,
  - low activity effluents consisting of floor runoff and resulting from circuit draining and from the cleaning of floors. There are also certain service wastes arising from laundry equipment, washbasins and showers.

Treatment

Hydrogenated primary effluents are fed to the treatment system which comprises:

- a filtration and demineralisation system intended to trap soluble and insoluble radioactive products,
- a degasing system intended to separate out any gases dissolved in the liquid phase,
- an evaporation system for separating the waste into a low activity distillate, which can if necessary be recycled, and a concentrate containing almost all the radioactivity (with the exception of tritium), substances in suspension and soluble salts (boron etc.).

The radioactive spent wastes are fed to treatment system TEU (spent waste treatment) where they undergo a treatment process adapted to their characteristics. The process tailings are treated in demineralisers. The chemical wastes are treated by

a process of evaporation. The floor runoff and the low activity service effluents (laundry, showers etc.) undergo filtration before being transferred into the tanks prior to release.

Release into Rivers or into the Sea

In the same way as for gaseous waste, liquid effluent wastes are subject to strict regulation which stipulates:

- the release conditions:
  - the total gamma radioactivity of the radioactive waste prior to release should be below 80 kBq/l,
  - the waste should be prediluted,
- annual limits (with tritium and excluding tritium),
- the nature of the monitoring to be carried out:
  - sampling downstream of the outfall point,
  - monitoring of groundwater,
  - analysis of sediments and aquatic vegetation.

3 - ASSESSMENT OF EFFLUENT RELEASES

3.1 - Gaseous Effluents

Figures 2 and 3 show, with an average per reactor, the releases of rare gases including tritium, and releases of halogens and aerosols, respectively.

Insofar as 900 MW plant units are concerned, the results are considered starting from 1982. The 1300 MW unit plants being of a later date (the first plant unit came on line in 1984 at Paluel), the values are counted from 1986 onwards.

Rare Gases and Tritium

Until 1986, the radioactivity of gaseous releases was determined by integrating the readings of the instrumentation installed in the stack. The declared radioactivity included, as a consequence, radioactivity actually released, plus virtual radioactivity owing to the "background" of the device (value indicated while the device is being traversed by air which is free of any trace of radioactivity). This method of accounting provided conservative measurements, but which were nevertheless far below the authorized annual limits.

In 1987, the operator was authorized to deduce by physical measurement the background of the device. This change in accounting procedure explains the discrepancy observed from this date. The releases are now at 4% of the authorized limit, taken as an average over the nuclear facilities.

Halogens and Aerosols

The values declared are determined from "beta gross" and "gamma gross" analyses carried out on the sampling devices for aerosols (filter paper) and halogens (active charcoal cartridges) respectively. The annual variations are slight. The releases correspond to less than 1% of the authorized annual limit taken as an average over the facilities.

3.2 - Liquid Waste

Figures 4 and 5 show, as an average for a given reactor, the tritium releases and the radioactivity releases excluding tritium.

Tritium

The annual amount of radioactivity released on balance, i.e. after two years of operation, was on average 12 TBq for the 900 MW units and 15 TBq for the 1300 MW units in 1991. In comparison with the electrical power generated, the releases of tritium were on average 2.7 Bq/Wh in 1991.

The stability of releases confirms that the production of tritium is directly proportional to the energy produced by the nuclear power plant. The tritium releases represent approximately 40% of the authorized annual limit.

Certain plant units using mixed plutonium and uranium oxide fuel (MOX), are prone to produce more tritium (theoretically 10%, owing mainly to ternary fission reactions of plutonium) and, therefore, to release more than a nuclear power plant utilizing uranium oxide fuel.

However, the monitoring procedure carried out by those nuclear power plants using MOX fuel has not demonstrated a noticeable increase in tritium radioactivity in the primary circuit water or in the effluents released.

Excluding Tritium

Changes to the regulations have led the operator to calculate radioactivity excluding tritium [A] (minus potassium 40 and radium) in the following manner:

. from 1983 to 1984 [A] =radioactivity (beta gross + 0.6 x gamma gross),

. from 1985 [A] =sum of the radioactivities of identified radionuclides

The main radionuclides identified in the liquid wastes are shown in Figure 6.

Cobalt-58 and 60, silver 110m and antimony predominate in liquid releases (> 80%) .

Since 1980, the releases of radioactivity excluding tritium

has been divided by 10. In 1991, they represent, as an average for a given reactor, 10 GBq/year or 2.5% of the authorized annual limit.

This result is the fruit of activities undertaken by EDF, since 1980, with the intention of reducing to a level as low as reasonably achievable (ALARA), the impact of nuclear power plant releases on the environment.

These activities have mainly concerned:

- the improvement of collecting circuits and of liquid effluent treatment (sorting waste according to its nature in order to avoid mixtures, and thereby facilitating further treatment and the accounting of the volumes produced, as well as installing means of supplementary treatment),
- the implementation of a system of organisation which makes it possible to strictly monitor both effluents and solid waste linked to the treatment process (motivation of personnel to limit the arising of waste at the source during maintenance operations, the creation of management committees authorized to set targets and to analyse any malfunctioning, and to designate a person in charge of coordinating site activities).

These efforts have made it possible to palpably reduce the volume of liquid effluents produced at the source (-50% for 900 MW units and -75% for 1300 MW units), as well as the volume of solid waste produced during treatment.

4 - CONCLUSION

The impact on the environment of liquid and gaseous releases from a nuclear site is very slight (the dose equivalent received annually by the most exposed member of the public is effectively several orders of magnitude below that which he receives from natural exposure).

Monitoring procedures, carried out at sites as part of environment monitoring, show that the earth's ecosystem is not noticeably effected by the releases from nuclear power plants. Artificial radioactivity (mainly cesium-134 and 137) measured by ground radiation meters results from fall-out from nuclear testing and from the Chernobyl accident; furthermore, it represents only a tiny fraction of natural radioactivity (potassium-40, thorium-238 etc.).

The aquatic ecosystem contains a few traces of the artificial radionuclides present in the liquid waste of nuclear power plants (cobalt-58 and 60, manganese-54 etc.).

Although from the point of view of health, any new action aiming to reduce releases would have little effect, efforts made up until the present day to improve the organisation and the proper management of effluents and solid waste will be continued, because they form part of the company strategy.

REFERENCES

[1] Liquid radioactive waste, its impact on the environment; Memorandum from EDF-SEPTEN, 69628 Villeurbanne Cedex.

[2] Arrêté du 10 août 1976, relating to the regulations concerning PWR nuclear power plants with regard to the release of their radioactive liquid effluents.

FIGURES

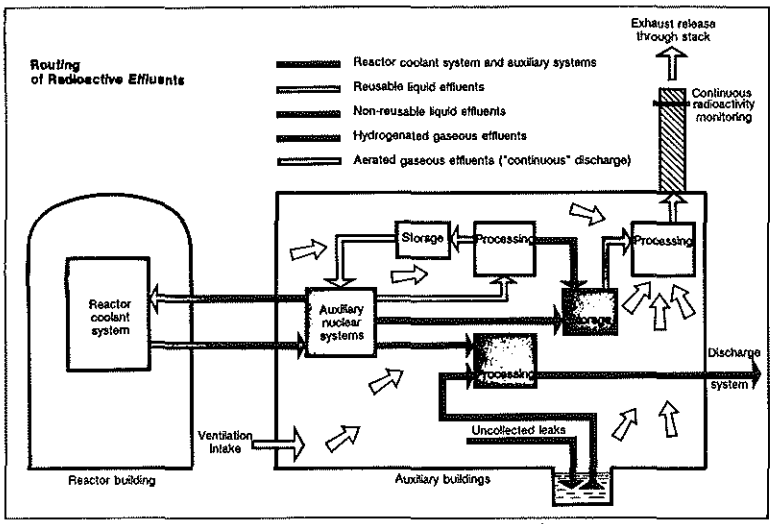


Figure 1

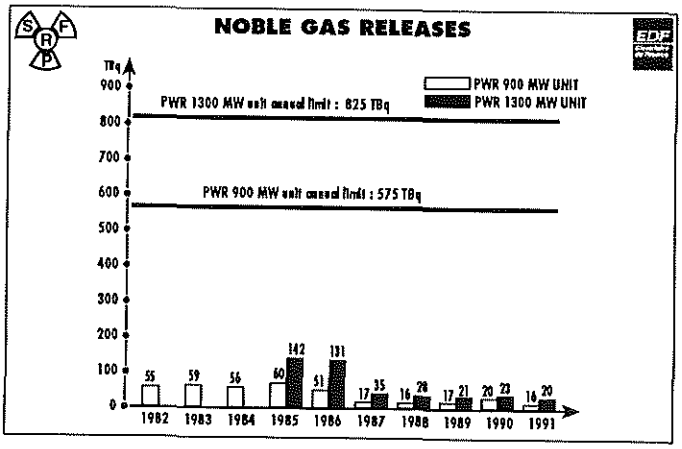


Figure 2

Figure 3

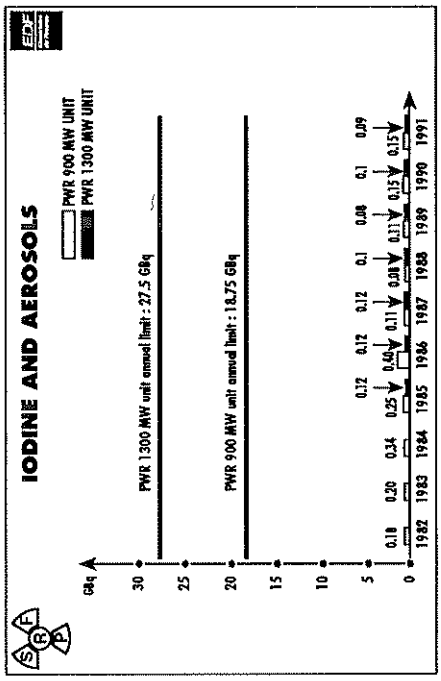


Figure 4

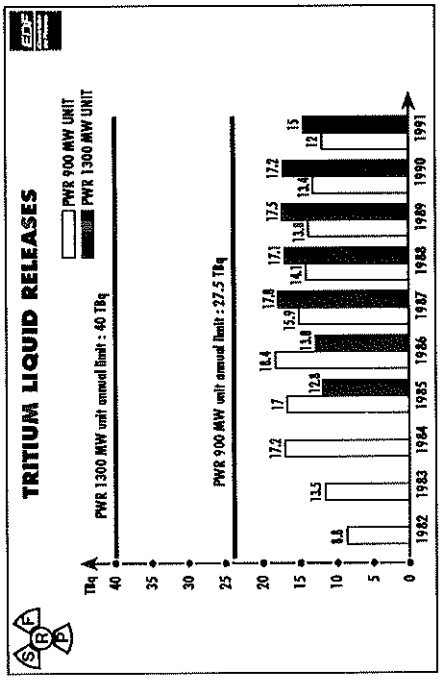


Figure 5

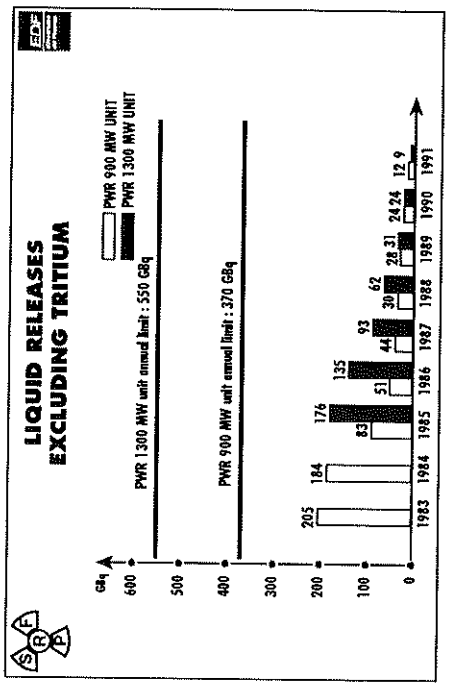
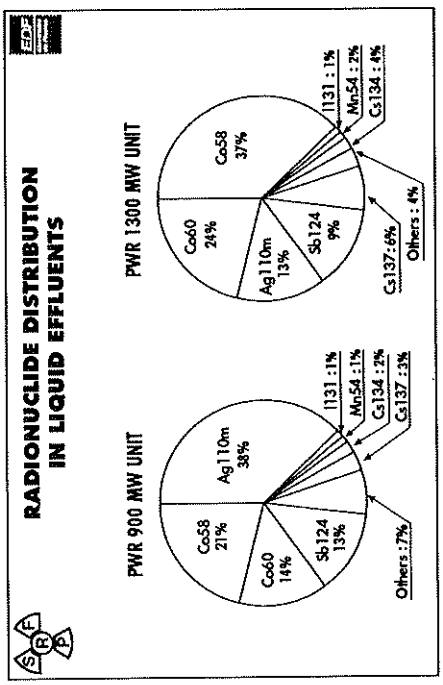


Figure 6



RESULTS OF THE WORK-RELATED PERSONAL DOSIMETRY AND THE SUPERVISION OF EMISSION IN A CONVOY PLANT

L. Izquierdo & U. Seyffer, Gemeinschaftskernkraftwerk Neckar, Postfach, D-7129 NECKARWESTHEIM

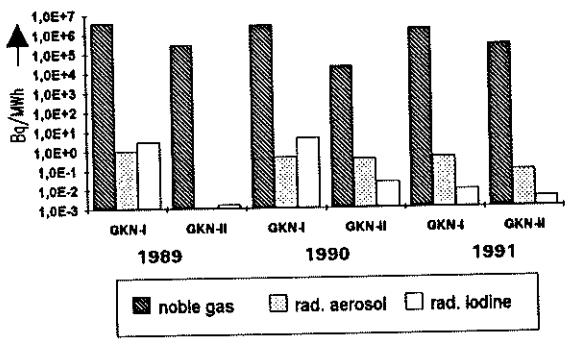
The third convoy pressurized water reactor, next to Emsland and Isar 2, the second unit of the Gemeinschaftskernkraftwerk Neckar GmbH, went into operation on 29th Dec. 1988 (1st criticality).

In this report some of the results gained sofar in a convoy plant are presented from the point of view of radiation protection. For better judgement the results from GKN II (1,356 MWe-4 loop-pressurized water reactor) are compared with those from GKN I (840 MWe-3 loop-pressurized water reactor) which has been in operation since the 23rd May 1976.

Supervision of emission

Here the emissions for the years 1989 - 1991 in the exhaust air und waste water from GKN I and GKN II were sorted according to nuclide groups and, for better comparison, were related to the quantity of electricity generated. As can be seen from the normalized exhaust air emissions in Fig. 1, the nuclide-group specific emissions are subject to fluctuations dictated mainly by the plant operations and to a lesser extent by the reactor type series.

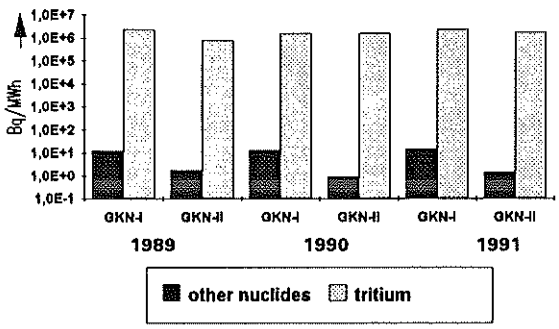
Fig. 1 Exhaust air emissions from the pressurized water reactor plants GKN I and GKN II related to the gross electrical energy generated



The aerosol emissions from GKN II are to be treated with caution because for organizational reasons, since 1990 reactor coolant system components from GKN I are brought into GKN II for decontamination and processing. Thus here a part of the activity embraces both units.

The same appllies to the waste water emissions (see Fig. 2), whereby tritium must be considered separately, since the tritium emissions in pressurized reactor plants are determined by the power dependent tritium generation from the boric acid and the quantity of cleaned but still tritium contamined waste water discharge.

Fig. 2 Waste water emissions from the pressurized water reactor plants GKN I and GKN II related to the gross electrical energy generated

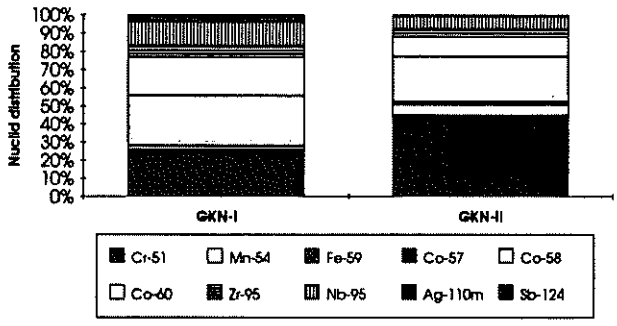


Here, due to the relatively long halftime of 12.3 years, an increase for GKN II is to be expected, whereas the emissions in GKN I have already reached a certain equilibrium.

Dose rate accumulation

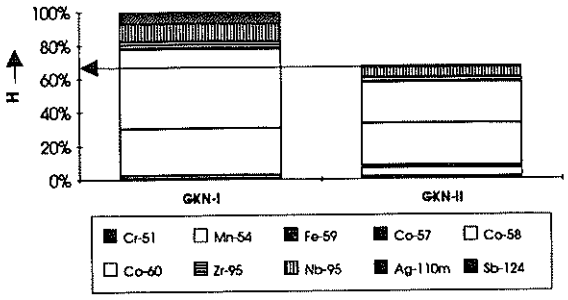
The relevant local dose rate for the radiation dose of personnel is determiend mainly by the activity inventory and its nuclide distribution. Since the inventory of free fission products depends on the integrity of the fuel assemblies, and this is influenced by several parameters not necessarily related to the plant type, the distribution of the corrosion products is thus considered here and compared. In order to be able to consider a nuclide distribution as wide as possible, particular value was put on the transportability of the nuclides. Thus direct measurements (insitu spectrometry) were waived and the distribution of the activity which can be wiped off was considered. For this purpose, during the inspections in 1991, nuclide adistributions gained from wipe tests were averaged and compared in percent (see Fig. 3a). As can be seen here the tracer nuclides in both plants are composed of the isotopes Cr-51, Co-58 and Co-60 (GKN I to approx. 75 % and GKN II to approx. 80 %).

Fig. 3a Percentual nuclide distribution of the corrosion products



While the procentual proportion of Co-58 resulting from the (n, γ) activation process from Ni-58 with 28 % in GKN I only differs marginally from the 25 % in GKN II, a clear shift to higher Cr-51 [(n, γ) activation from Co-50] and lower Co-60 [(n, γ) activation from Co-59], [n, p) from Ni-60)] proportions can be recognized. This difference make a difference in the radiological evaluation.

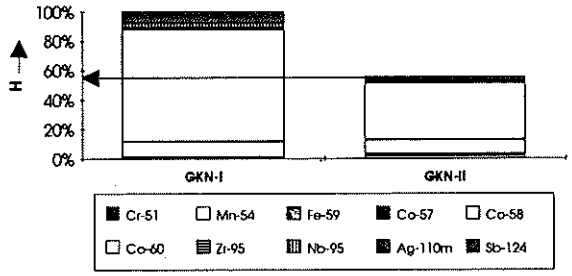
Fig. 3b Nuclide distribution of the corrosion products; normalized to the dose rate at a distance of 1,000 mm



For better evaluation, in Fig. 3 b the nuclide distribution has been normalized to the dose rate. Here dose rate proportions have been calculated for an aera of infinite extent. Due to the high proportion of Cr-51, resulting from its low dose rate constant, which contributes little to the dose rate accumulation, the dose rate in GKN II is reduced to 67 % of the GKN I value. If one considers this model with an additional 60 mm thick steel shielding (this corresponds to the wall thickness of primary circuit pipes in pressurized water reactors), the dose rate is reduced to the low Mn-54 and Co-58 contribution of 56 % according to Fig. 3 c.

Although the Co-60 proportions have not yet reached equilibrium in GKN II, due to the use of lowcobalt or cobalt-free steels a significant reduction of the Co-60 proportions from the (n, γ) nuclear reaction of the Co-59 is recognizable, which leaves hope that the generally low annual accumulative doses can be maintained.

Fig. 3c Nuclid distribution of the corrosion products; normalized to the dose rate at a distance of 1,000 mm with a 60 mm steel shielding



Supervision of personnel and accumulative doses

Between 1976 and 1989 pen dosimeters were used at GKN as immediately readable (inofficial) dosimeters for supervision of the personal doses. For all work in the controlled area, a work order is necessary. The necessary radiation protection measures for this work are defined in the socalled radiation protection notice.

The dose values were entered into a computer and stored in the work related dose record, parallel to the personal dose record. Thereby it was possible at any time to extract the dose level for the work and compare it with either the planned values or the doses of individual employees, groups of persons or external contractors.

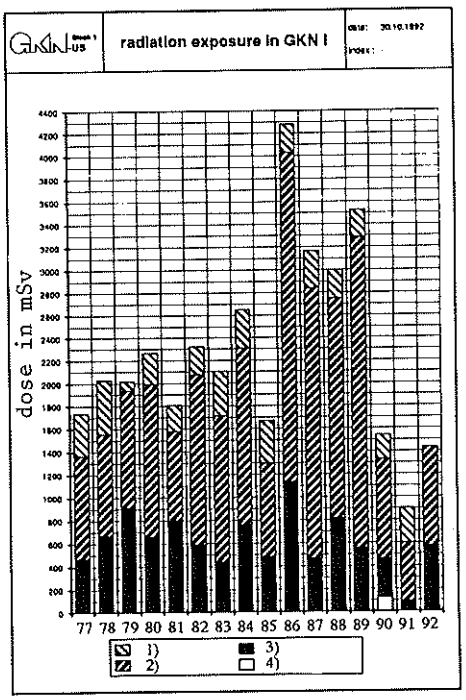
Since 1990 the immediately readable dosimeter used is the electronic dosimeter system RAD90 form the Alnor company, Herfurth. At the entrance to the controlled area, the allocation of the electronic dosimeter to the person is done by means of an identity card. An allocation of the person to a work order is carried out by means of a manual entry by the radiation protection personnel at the gate to the controlled area.

At the exit of the controlled area the dose ist automatically read out (revolving gate and dosimeter readers) and allocated to the personal dose record and the work order record.

Then by EDP an individual, extensive evaluation is possible.

This and the change in the procedure for immediately readable dosimeters has the consequence that the dose values even for comparable jobs cannot be exactly compared with one another.

The dose values given in the following are always related to the dose determined by the immediately readable dosimeters.



Dose trends for GKN I and GKN II

Fig. 5 shows the trend of the radiation exposure in GKN I for the period 1977 - 1991.

There is not a typical trend but rather the accumulative dose varies very greatly. This is due to the often extensive and doseextensive repair, refitting and modification work.

1) annual
2) inspection outage dose
3) non destructive testing
4) dose from GNK I in GKN II

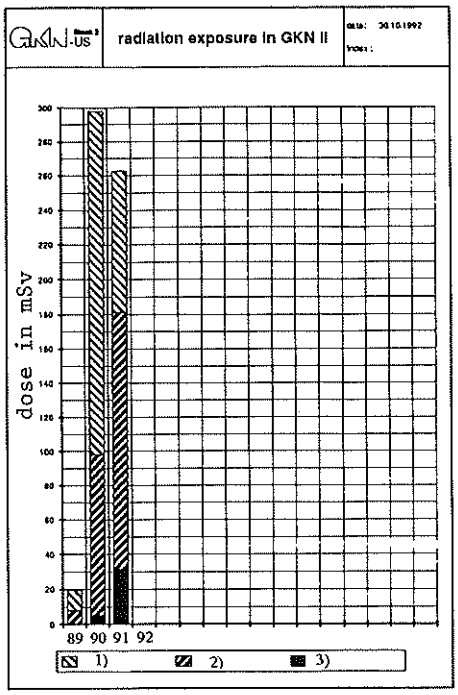


Fig. 6 shows the radiation exposure in GKN II. These values are significantly lower. One reason is that, at least up to now, still no additional work comparable to that in GKN I has been necessary.

1) annual dose
2) inspection outage dose
3) non destructive dose

Dose comparison for GKN I and GKN II

The fact that the radiation exposure in GKN II is significantly lower can be explained by the following points:

1. Already in the planning and layout of the plant, radiation protection aspects were considered.

2. Use of materials which do not, or only very slightly, contribute to activation. Thus also the level of the reactor coolant system activity is less.

GKN I:	Cr-51	ca. 2 E 6 Bq/Mg		**GKN II:**	Cr-51	ca. 2 E 6 Bq/Mg
	Co-58	ca. 2 E 6 Bq/Mg			Co-58	ca. 2 E 6 Bq/Mg
	Co-60	ca. 1 E 6 Bq/Mg			Co-60	ca. 2 E 5 Bq/Mg
	Sb-124	ca. 2 E 6 Bq/Mg			Sb-126	ca. 2 E 5 Bq/MG

3. Basicly safe components and aggregates which are easy to service and repair.

4. Accessibility, arrangement of the components, spacial separation.

5. Arrangement such that no additional scaffolding or working platforms are necessary.

6. Rapidly assemblable insulations.

7. The tracks and mounting points for mechanized tests were taken into account in the planning and istalled during the building phase of the plant.

8. Additional construction aids, cranes and elevators, crossheads, handling facilities, crane tracks etc.

9. Storage and standing areas.

10. Permanently fixed decontamination facilities (eg. for reactor coolant pumps).

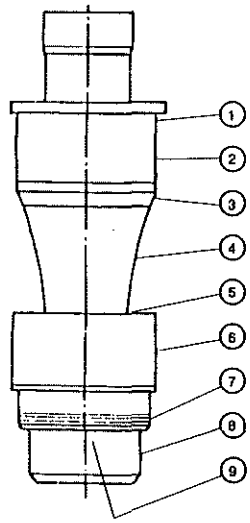
11. Continous development of the equipment technology.

12. Consequent application of experience gained from existing plants.

Dose rate comparison GKN I and GKN II

GKN-US

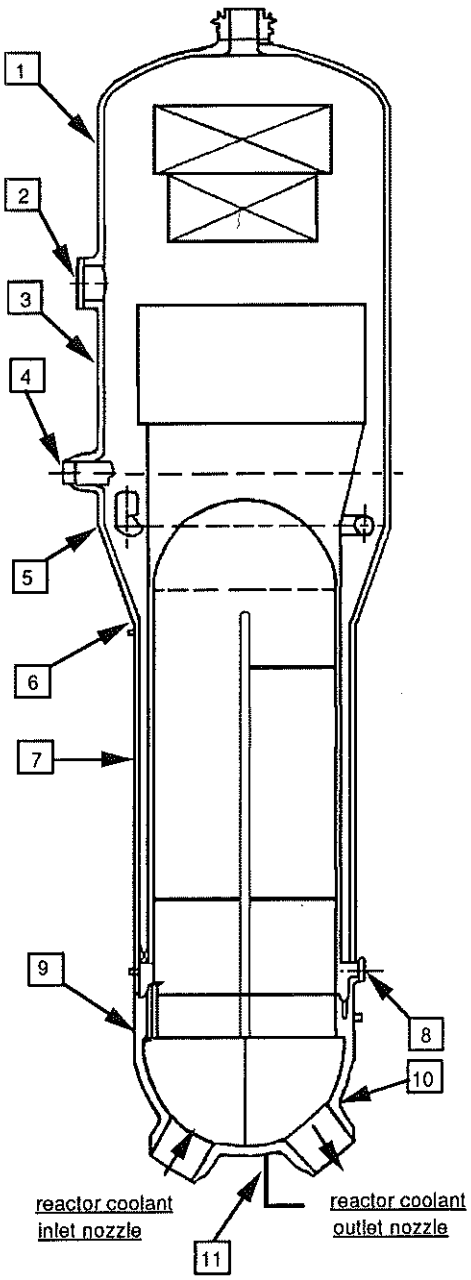
reactor coolant pump



measure point	$\dot{H}$ in mSv/h GKN I YD 10 D 001	GKN II JEB 40 AP 001
1	120	9,0
2	130	9,5
3	250	13,5
4	300	14,0
5	700	26,0
6	200	10,2
7	250	19,5
8	230	15,0
9	600	19,0

GKN-US

steam generator



steam generator filled

	GKN I YB10B001	GKN II JEA40BC001
measure point	$\dot{H}$	(µSv/h)
	contact	contact
1	< 10	<1
2	< 10	<1
3	< 10	<1
4	< 10	<1
5	50	1
6	150	15
7	300	10
8	200	5
9	500	10
10	150	10
11	2500	1500
Datum	28.07.89	03.09.92

Accumulative doses

The following figures show in a few exemples the accumulative doses for particular jobs. It can be seen that despite the partly high personnel requirement, the accumulative doses are very low.

For example during the GKN inspection in 1992 up to 330 persons were simultaneously in the controlled area.

The accumulative doses of the other convoy plants are of the same order of magnitude. A global comparison is however not sensible for the previously mentioned reasons. (Method of dose determination, extent of work, comparability of the individual jobs. It is emphatically advised that for a detailed comparison, all conditions should be very carefully analyzed.

Conclusions and outlook

Our experience up to now shows that with the convoy plants a high level has been archieved with respect to low activity memissions and low accumulative doses.

Thereby, the activity emission is dependent on the concentration of the fission products (fuel assembly defective yes/no) and on the work carried out.

The accumulatiuve doses result from the routine work (service, maintenance, repetitive inspections, plant operation, changing of fuel assemblies etc.) and the repair, modification and refitting works. Through the plant and systems technology very good preconditions for optimum redation protection have been achieved. Also, the low accumulative doses will be maintained for routine work despite perhaps a slight increase of the level of the local dose rate.

The total accumulative dose then depends on the additional other work. Thereby the refitting and modification works are of course of very great significance. These must thus be examined (point of time and necessity) whether the dose to be expected is justified.

GKN-US

Comparation of dose by identifical works

dose in mSv	1989		1990		1991		1992	
	GKN I	GKN II	GKN I	GKN II	GKN I	GKN II	GKN I	GKN II
annual dose	3531	19,7	1548	298,0	908	263,0	---	---
inspection outage dose	3288	8,0	1336	98,4	600	181,7	1440	---
fuel changing	121	---	117	9,5	91	24,5	63,9	---
non destructive testing	557	---	462	5,4	95	31,7	575	---
health physics	141	2,1	81	9,0	63	22,9	---	---
decontamination and waste treatment	329	2,1	167	18,5	100	66,9 *	---	---

* incl. reactor coolant pump review from GKN I

RADIOACTIVITY DISCHARGES FROM SWISS NUCLEAR POWER STATIONS AND THEIR RADIOLOGICAL IMPACT

Hansruedi Völkle [1], Franz Cartier [2] and Christophe Murith [1]

[1] Sektion Ueberwachung der Radioaktivität (SUER), Federal Office of Public Health (BAG), Chemin du Musée 3, CH-1700 FRIBOURG/SWITZERLAND

[2] Nuclear Safety Inspectorate (HSK), CH-5232 VILLIGEN-HSK/SWITZERLAND

Summary

Radioactivity discharges from Swiss Nuclear Power Stations (NPS) to the environment and the radiation exposures of the surrounding population as well as those of the NPS workers have decreased slightly in the last 10 years due to continuous technical improvement of plant and nuclear fuel. The effective doses of persons living near Swiss NPS are for conservative assumptions at least one order of magnitude below the conceptual dose limit of 0.2 mSv/year. The principles of the release monitoring programme, the population exposure calculations and the results are presented and discussed.

1. Technical Data on Swiss Nuclear Power Stations

Switzerland has five NPS on the four sites Beznau (KKB), Mühleberg (KKM), Gösgen-Däniken (KKG) and Leibstadt (KKL):

Table 1: Technical Data.

Data	KKB I+II Beznau	KKM Mühleberg	KKG Gösgen	KKL Leibstadt
MW_{th} (nominal)	2 x 1130	997	2900	3138
MW_{el} (nominal)	2 x 364	336	970	1045
MW_{el} (net)	2 x 350	322	920	990
Reactor Type	2 x PWR	BWR	PWR	BWR
Reactor Manufacturer	Westinghouse	GE	KWU	GE
Turbine Manufacturer	BBC	BBC	KWU	BBC
Generators(s) (MVA)	4 x 228	2 x 190	1 x 1140	1 x 1180
Cooling Medium	river Aare	river Aare	atmosphere wet tower	atmosphere wet tower
Commercial Operation since:	1969/1971	1972	1979	1984
Total TW_{el} (net) produced until end 1991:	I=53.14 II=52.81	46.15	84.81	52.53

2. Radiation Exposure of Swiss NPS Workers

The collective effective doses (external whole body irradiation) of the Swiss NPS workers including external staff are given in figure 1. The number of workers (without external staff) in 1991 was: KKB I+II: 371; KKG: 270; KKL: 329 and KKM: 225. The mean annual effective dose decreased from 5.2 mSv/year in 1986 to 2.1 mSv/year in 1991 and the fraction of workers exceeding 20 mSv/year dropped in the same period from 7 % to 0.3 %; no one had effective doses over 50 mSv/year. (see also [2])

3. Control of Radioactivity Discharges to the Environment

The Nuclear Safety Inspectorate (HSK), the licensing authority has limited the maximum radioactivity discharges to the environment in such a way, that no one of the surrounding population could accumulate more than 0.2 mSv/year by incorporation of contaminated food, water and breathing air and by external exposure. Direct ^{16}N radiation from the turbine building should not cause ambient doses of more than 0.1 mSv/week outside the fenced NPS area.

This leads for radioactivity discharges to the environment by the exhaust air trough the stack to annual and daily limits for the weighted sum of radioactive noble gases, annual and weekly limits for radioactive ^{131}iodine, annual and weekly limits for radioactive aerosols with half live longer than 8 days. For the discharges by waste water into the river there are annual limits for tritium, annual limits for the weighted sum of all other radionuclides and finally an activity concentration limit for the weighted sum of the radionuclides in waste water discharged to the river.

Airborne radioactivity releases are controlled by continuous gamma, total gamma respectively total beta measurements (for iodine, noble gases or aerosols). The isotopic composition of the exhaust air is determined by daily air samples (for noble gases), weekly aerosol and charcoal samples (for aerosols and iodine) by gamma spectrometric measurements. Total alpha and $^{89}Sr/^{90}Sr$ are determined on quarterly composite aerosols samples. ^{14}C in the exhaust air is measured on request of the licensing authority. For radioactivity in waste water, samples from the tank to be released are measured for the discharge balance and to decide whether the tank can be evacuated to the river or the water has to be treated furthermore by evaporation or by ion exchange resins. These measurements are carried out by the NPS laboratory and are controlled by additional samples and measurements performed by the HSK and SUER Laboratories. (see also [2]).

4. Models used for Calculation of the Radiological Impact

On the basis of the annual radioactivity discharge balance established by the NPP, controlled by HSK, the lather calculates the overall radiation exposure of the population in the vicinity of the plant. These calculations are essentially based on the German AVV-Regulations (presented later in this meeting), in some points adopted to Swiss habits, and are most conservative, i.e. are giving effective doses for the worst case in concern to living and nutrition habits of the population (see [3]). The real radiation exposure values of persons living near Swiss NPS may be lower by one order of magnitude ore even more. (see also [2]).

5. Radioactivity Discharges from Swiss NPS since 1970 and Calculated Radiation Exposure to the Surrounding Population.

The annual values for noble gas, ^{131}iodine, aerosols (>8 day of half live), tritium and other nuclides are presented in Fig. 2 - 6. The values for noble gas discharges are given in ^{133}Xe equivalent, i.e. the weighted sum of all noble gases using weighting factors given by the toxicity ratios between each noble gas isotope and ^{133}Xe. The gaseous ^{14}carbon discharges in GBq/year as carbon dioxide and organically bound carbon and their corresponding population radiation exposures (values in parentheses for adults/children in μSv/year) are: KKB I+II: 40 (1.1/5.3); KKM: 200 (0.68/3.2); KKG: 100 (0.68/3.2); KKL: 340 (1.8/8.3). Apart from ^{14}carbon and tritium all other radioactive discharges to the environment from modern NPS can bee kept extremely low by using ion exchangers and distillation for waste water, delay lines and active charcoal filter for noble gases and dust filters for radioactive aerosols, together with other technical improvements like increased fuel quality and the use of low cobalt steel. This leads to a slowly decreasing trend for the calculated (conservative) radiation exposure for the

surrounding population as shown in Fig. 7 from about 10 - 100 μSv/year in the seventies to about 0.1 - 1 μSv/year in the nineties (The 1986 incident in KKM will be discussed in the section 6). Therefore the relative contribution of ^{14}carbon to the population radiation exposure (values for adults for 1991) is quite high: KKB I+II: 62%; KKM: 4.2%; KKG: 94%; KKL: 91%. (see also [2]).

6. Incidents

Apart from the Lucens incident in January 1969 (a small experimental reactor) no severe safety relevant incidents or accidents occurred in commercial Swiss NPS. All events were anomalies according to degree 1 or lower of the IAEA International Nuclear Event Scale. One such anomaly was the KKM event of September 1986 where radioactive resins (used to treat waste water) were released uncontrolled by the ventilation air trough the stack. The total activity release was near the annual limit for radioactive aerosols and the calculated radiation exposure of a farmer family living some 500 m west in the most charged region was near the annual dose limit of 0.2 mSv (see also [1]).

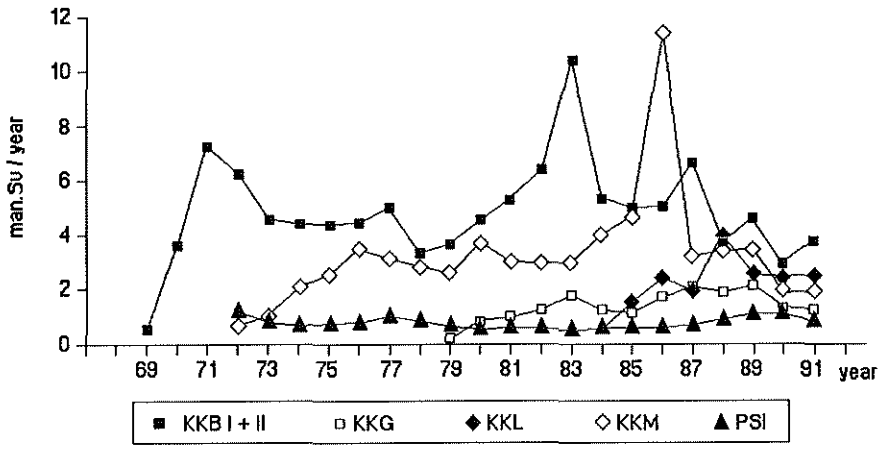
7. Literature

[1] J. Czarnecki: "Some Experiences with measurements of stack releases and their correlation with environmental measurements". Proceedings IRPA-7, 10-17 April 1988, Sidney, pp.689.

[2] Annual reports of the Swiss Nuclear Safety Inspectorate (HSK) and the annual reports on environmental radioactivity issued by the Federal Office of Public Health (BAG) in Bern (former KUER reports).

[3] A. Leupin & F. Cartier: "Emission Limits and Doses in the Vicinity of a Nuclear Facility Having several Points of Release for the Example of the PSI"; in this symposium.

Fig. 1 Annual Collective Effective Doses to Workers (including external staff) in Swiss NPS and in the Paul Scherer Institute (PSI) in man.Su / year



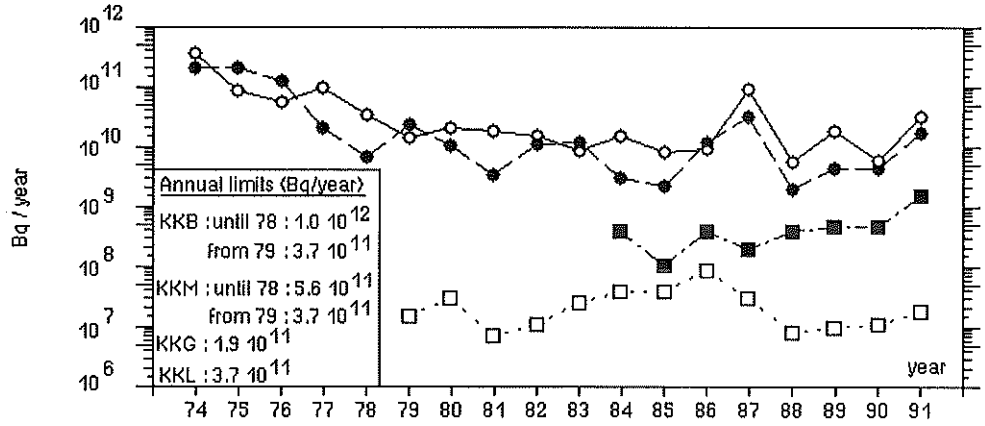


Fig.2 Liquid Radioactivity Discharges : Waste Water (without Tritium) : Bq/year

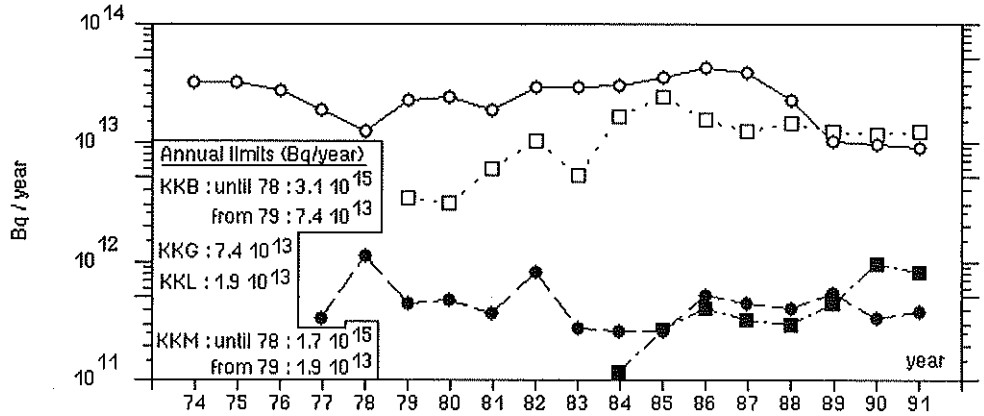


Fig.3 Liquid Radioactivity Discharges : Tritium : Bq/year

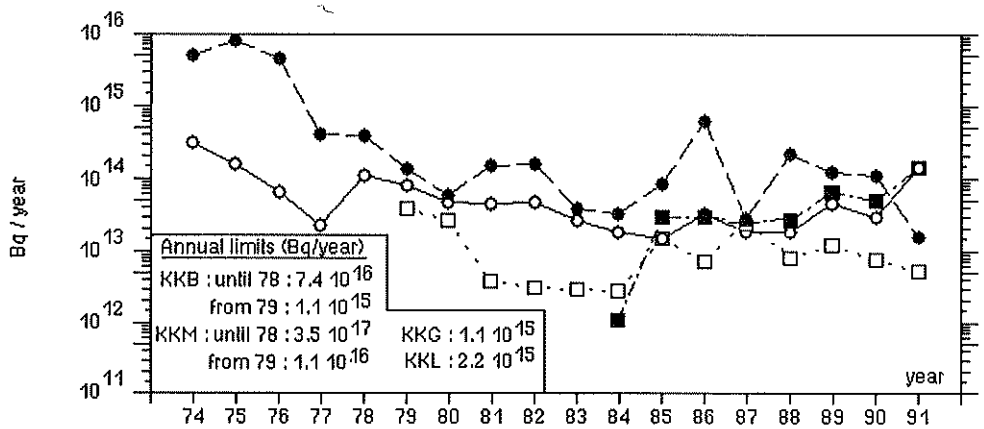


Fig.4 Gaseous Radioactivity Discharges : Noble gases (Bq/year Xe-133-equivalent)

Swiss Nuclear Power Stations		
	KKB (PWR; since 1969/71)	KKG (PWR; since 1979)
	KKM (BWR; since 1972)	KKL (BWR; since 1984)

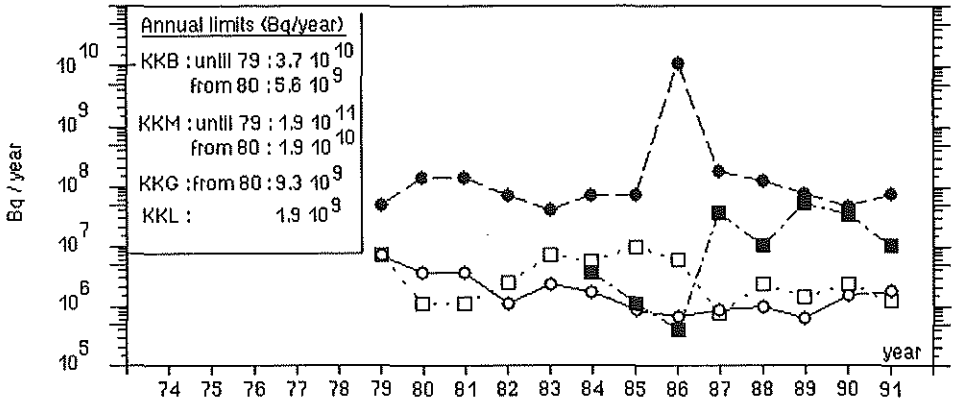


Fig.5 Gaseous Radioactivity Discharges : radioactive Aerosols (T1/2>8d) Bq/year

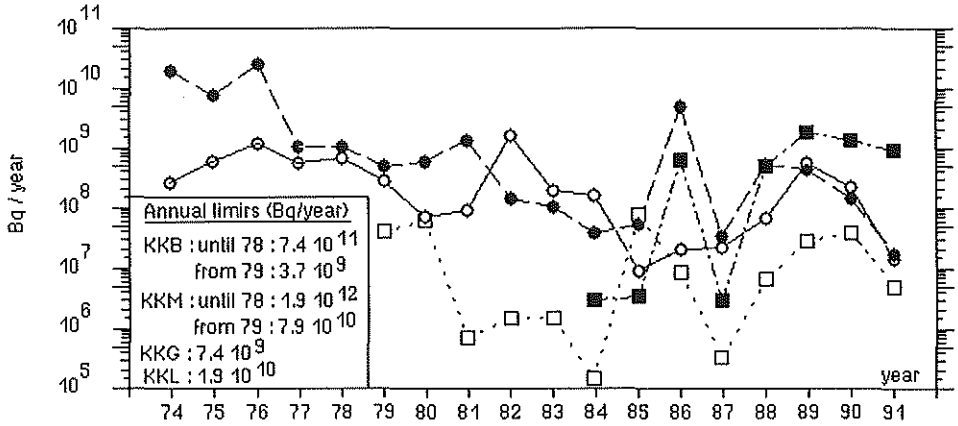


Fig.6 Gaseous Radioactivity Discharges : elementary Iodine-131 (Bq/year)

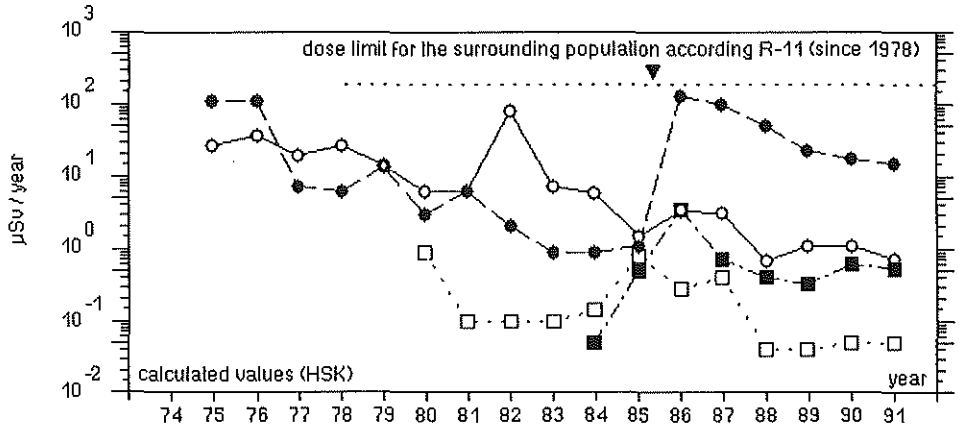


Fig.7 Calculated Maximum Population Radiation Exposure in µSv/year

Swiss Nuclear Power Stations		
	KKB (PWR; since 1969/71)	KKG (PWR; since 1979)
	KKM (BWR; since 1972)	KKL (BWR; since 1984)

MONITORING OF LIQUID RADIOACTIVE EFFLUENTS OF NUCLEAR POWER STATIONS IN THE FEDERAL REPUBLIC OF GERMANY

H. Rühle[1)] and I. Gans[2)]
[1)] Institute for Water, Soil and Air Hygiene of the Federal Health Office
[2)] State Office of the Environment and Nature Mecklenburg-Vorpommern

Summary

Measures and regulations for monitoring liquid radioactive effluents of nuclear power stations are specified in a Rule of the "Kerntechnischer Ausschuß (KTA)" (Nuclear Technical Committee). The total emissions of radioactive substances in liquid effluents in the Federal Republic of Germany show a significantly decreasing tendency during the period of 1980 to 1990 while the number of nuclear stations in Germany increased during this period. The very small amounts of radioactivity which, due to the available technology, are released to the environment result in very low radiation exposures of population via aquatic exposure pathways.

1. Existing regulatory framework

In 1969, the Minister of Education und Science requested the Institute of Water, Soil and Air Hygiene of the Federal Health Office (WaBoLu), on the basis of a research contract, to develop and test methods for monitoring of liquid effluents.

The result of this research work were used as a basis for the "Rules for Measurements and Control of Discharges of Radioactive Liquid Effluents from Nuclear Power Stations with Light Water Reactors" which where approved by the "Länderausschuß für Atomkernenergie" (State Committee an Atomic Energy) in 1972.

Further experiences which were gained by WaBoLu in the course of monitoring liquid radioactive effluents of the nuclear power stations operating at that time, were incorporated in "Sicherheitstechnische Regel 1504" (Safety Rule 1504) of the "Kerntechnischer Ausschuß" (Nuclear Technical Committee) published in 1978 after it had been drafted jointly by "Kerntechnischer Ausschuß" and "Länderarbeitsgemeinschaft Wasser (LAWA)" (State Working Group Water) [1]. This rule provides measures und measurements which have to be respected and performed by the operators in order to monitor and evaluate the discharges of liquid radioactive effluents from nuclear power stations.

In the same year (1978), the "Länderausschuß für Atomkernenergie" approved the "Guidelines for the Control of the Operators Monitoring of Radioactive Emissions from Nuclear Power Stations", in order to assure a sufficient quality control of the measurements of the operators by officially appointed experts [2]. According to these guidelines the officially appointed expert has to perform a number of measurements parallel to those of the operator performed for balancing the discharge of radionuclides. Additionally, operator and official expert have to take part in a quality assurance program which, amongst other measures, includes participation in an interlaboratory control test. The latter is run every year by the Federal Health Office in cooperation with the Physikalisch-Technische Bundesanstalt.

2. Measurements for balancing the discharge of radionuclides

In the nuclear power stations, after the different steps of treatment the liquid effluents are collected prior to release in specially designed discharge tanks. Measures which have to be taken by the operator for monitoring radioactive liquid effluents in accordance with the Rule 1504 of the KTA, are listed in table 1.

Table 1. Measures for monitoring liquid radioactive effluents according to Rule 1504 of the KTA

LIQUID EFFLUENTS (Discharge tanks)
1. Sampling Sample representative for the total volume of the discharge tank, 1 l for "decision measurement", 1 l stored for long-term evidence. Mixing of composite samples (weekly, monthly, quarterly)
2. Measurement for decision on discharge Gross γ-measurement of a sample (volume 1 l) (calibration by Cs 137) Concentration limit for discharges: 1.9 E07 Bq/m^3 (5 E-04 Ci/m^3) Discharge from a tank only after written approval of an authorized staff member
3. Continuous monitoring of the discharge In the discharge pipe device for gross γ-measurement (calibration by Cs 137) Concentration limit for interrupting discharge: 1.9 E07 Bq/m^3 (5 E-04 Ci/m^3).
4. Measurements for balancing discharges γ-spectrometry on weekly composite samples, Sr 89/90 in monthly composite samples, tritium and gross-α-activity in quarterly composite samples - deadlines for measurements provided *) - lower limits of detection for measurements provided
5. Storage for long-term evidence Sample of 1 l of each discharge to be stored for 1 year.
*) deadlines for measurements in order to balance discharges γ-spectrometry: within the week following sampling period other measurements: within one month after end of sampling period

Liquid effluents may only be discharged after the specific measurement for decision on discharge has proved that the concentration limit of 1.9 E07 Bq/m^3 (5 E-04 Ci/m^3) has not been passed. For the purpose of balancing the radioactive releases from the discharge tanks, γ-emitters have to be analysed in weekly composite samples, strontium 89 and strontium 90 in monthly composite samples and tritium and gross-α-activity in quarterly samples. For all measurements deadlines for the performance and lower limits of detections are provided. Finally, samples of each discharge have to be stored in order to provide evidence for possibly necessary analyses at a later time.

3. The emission information system (EMIS) at the WaBoLu

For the evaluation and documentation of the release of liquid radioactive effluents from nuclear power stations, the emission information system EMIS has been installed since 1970 [3,4]. In its data bank all data which are relevant to the measurements performed by the operator for balancing the discharges (see 2.) are stored. These data are the concentrations of γ-emitters, strontium isotopes, tritium and gross-α-activity, as well as the volume of discharged effluents on a weekly basis. If radionuclides are not detected (e.g. by the γ-spectrometric measurement 22 radionuclides are specifically investigated), the lower limit of detection reached in the specific measurement is documented. Computer programs are provided for calculation of discharge balances for certain periods of time, e.g. weeks, months, quarters or years, on this data base.

4. Documentation of special measurement sequences of liquid effluent monitoring

In the following part a few samples for the use of the emission information system will be given.

Figures 1 and 2 show weekly discharges of the nuclear power station Philippsburg 2 in 1990. In figure 1 the total discharge of fission and activation products (except tritium) and the discharge of Co 60 for each week of the year are displayed. The elevated discharges during the weeks 6 to 12 (February/March) were caused by the routine maintenance and refuelling period of the station Unit 2. The slightly elevated discharges during the weeks 23 to 29 (June/July) result from laundry effluents from Unit 1 of the plant, which had its maintenance and refuelling period during these weeks. Parts of its contaminated waste water were transferred to Unit 2 for decontamination and increased the normal level of discharges. Figure 2 shows a similar time sequence for the radionuclides Cs 134 and Cs 137.

Figure 3 gives an example for discharges which show elevated levels over a short period of time at the nuclear power station Biblis Unit A in 1990. The relatively high discharges during the weeks 6 and 7 were caused by the fact that during these weeks the evaparator used normally for effluent treatment was out of operation for maintenance reasons. The effluents were decontaminated by filtration during this time. The figure shows that the use of an evaporator leads to a decontamination which is better by factor of 50 compared to filtration. The elevated release levels during the weeks 24 to 32 (June - August) are due to the routine maintenance and refuelling period.

5. Development of total releases of radioactivity in liquid effluents in the Federal Republic of Germany

According to the principle of radiation protection in order to keep the release of radioactive substances into the environment as low as possible, and taking into account the state of science and technology, measures for decreasing the discharges of liquid effluents from nuclear power stations have steadily been improved in the past. Predominant reasons for the reduction of the discharges are improvements in fuel fabrication, improved treatment methods for the effluents, and optimized strategies for the use of these methods due to the increased experience of the operators staff.

Figure 4 shows the sum of the yearly releases of fission and corrosion products of all nuclear power stations in the Federal Republic of Germany from 1980 to 1990 (for boiling (BWR) and pressurized water reactors (PWR) separately). The decreasing tendency gains additional importance by the fact that after 1980 3 power stations with BWR and 9 stations with PWR started operating. The total emissions of fission and corrosion products (without tritium) decreased from about 50 GBq in 1980 to less then 5 GBq in 1990, i.e. they were reduced by a factor of 10. Figure 5 shows the same tendency for the different rivers receiving discharges from nuclear power stations.

This tendency of reduction of discharges is not observed in the case of tritium which cannot be removed by effluent treatment. Figure 6 shows the total release of tritium by boiling and pressurized water reactors from 1980 to 1990. Until 1985 the releases of PWRs increased due to the increase in operating facilities. During the last years the total discharge remained almost constant and was about 140 GBq. The releases from BWRs are unimportant compared to PWRs.

The release of fission and corrosion products are very low compared to the authorized discharge limits. The resulting radiation exposure of population due to the releases of liquid radioactive effluents is very low and of the order of 1 µSv/a and less. The calculated values are published by the Minister of Environment, Nature Conservation and Reactor Safety in yearly reports.

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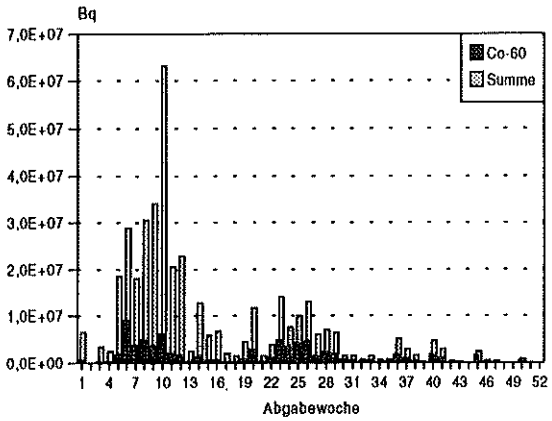


Abbildung 1: Kraftwerk Philippsburg 2
Abgabe radioaktiver Stoffe mit dem Abwasser (1990)

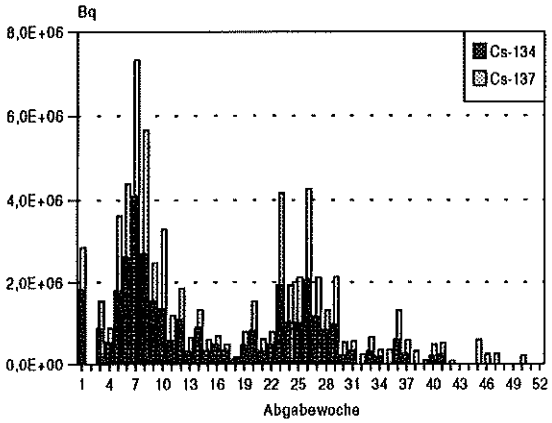


Abbildung 2: Kraftwerk Philippsburg 2
Abgabe radioaktiver Stoffe mit dem Abwasser (1990)

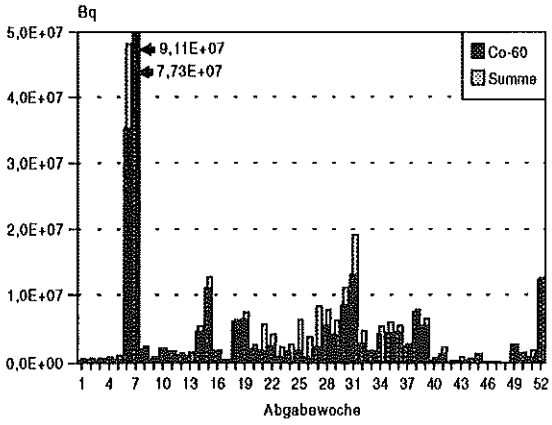


Abbildung 3: Kraftwerk Biblis A
Abgabe radioaktiver Stoffe mit dem Abwasser (1990)

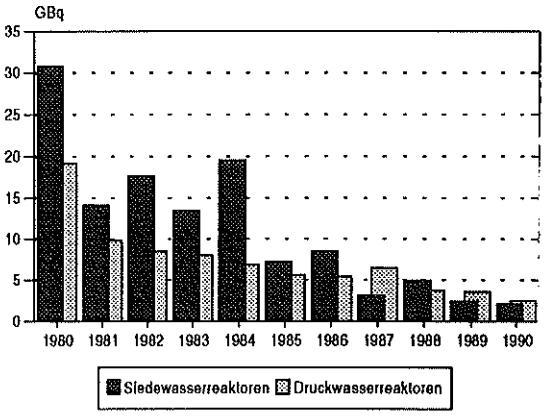


Abbildung 4: Abgabe radioaktiver Stoffe mit dem Abwasser aus Kernkraftwerken in der Bundesrepublik Deutschland Spalt- und Aktivierungsprodukte (außer Tritium)

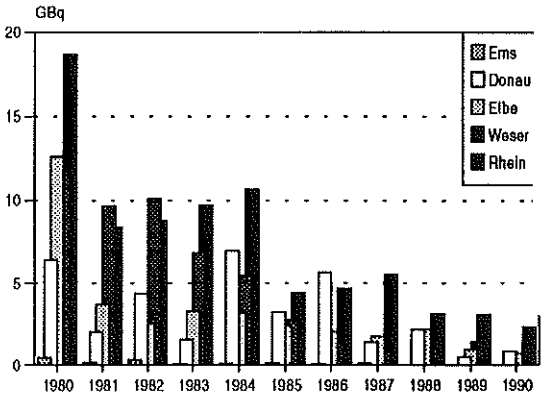


Abbildung 5: Abgabe radioaktiver Stoffe mit dem Abwasser aus Kernkraftwerken in der Bundesrepublik Deutschland Spalt- und Aktivierungsprodukte (ohne Tritium)

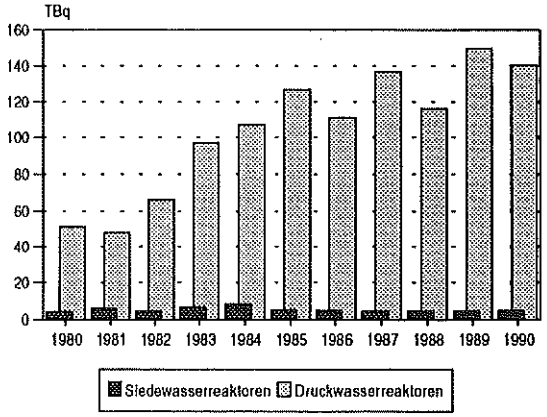


Abbildung 6: Abgabe radioaktiver Stoffe mit dem Abwasser aus Kernkraftwerken in der Bundesrepublik Deutschland Tritium

IMPACT OF THE *LA HAGUE* REPROCESSING PLANT ON THE SURROUNDING ENVIRONMENT

J.BETIS, Adjoint au Directeur
Sûreté-Qualité-Méthodes
de l'Etablissement de COGEMA La Hague

1. INTRODUCTION

The La Hague plant was designed and constructed to reprocess spent fuel; its current capacity of 1200 metric tons per year will rise to 1600 tons per year by 1994. One of the principal concerns of nuclear operators is to check that radiation protection standards are applied, and especially to control the impact on environment.

2. REGULATORY REQUIREMENTS

Before start-up of a nuclear facility, the impacts of facility operations on the surrounding environment and, more specifically, on the public must be carefully assessed. This assessment addresses the health effects of plant operations under both normal and accidental conditions, and is the basis for official French authorities, which establish release limits to be respected at all times. The regulations concern liquid and gaseous releases, each of which have four activity thresholds for specific elements:

liquid releases: tritium, beta emitters (excluding tritium), strontium 90 and cesium 137, and alpha emitters;

gaseous releases: gases other than tritium, tritium, halogens, and aerosols.

Annual release authorization			
LIQUID RELEASES		**GASEOUS RELEASES**	
Tritium	37.000 TBq	Gases (other than tritium)	480.000 TBq
Beta emitters (excluding tritium)	1700 TBq	Tritium	2200 TBq
90 Sr and 137 Cs	220 TBq	Halogens	110 GBq
Alpha emitters	1,7 TBq	Aerosols	74 GBq

Compliance with the release limits is checked through radiological monitoring of the site and its environment. All results of radiological measurements which trace the evolution of the environment are sent to be checked to the Service Central de Protection contre les Rayonnements Ionisants (SCPRI), or Central Radiation Protection Service, of the Ministry of Health.

3.THE MONITORING PROGRAM

The methods used for monitoring and surveillance may be divided into two different categories with respect to their implementation:

continuous monitoring, which rapidly activates necessary corrective actions in the event of changes in a monitored parameter; due to the large number of parameters involved, monitoring is conducted from a central control room at La Hague in an environment control room.

delayed monitoring, using environmental sampling and laboratory analysis, to add further data to that gathered by continuous monitoring and dosimetry.

Continuous Monitoring

Real-time monitoring is performed on gaseous releases, drainage systems and meteorological parameters.

Gaseous Releases

Gaseous releases are monitored continuously on three concentric circles starting with the plant. The inner circle centers on the main stacks of the UP2, UP2 800, and UP3 plants. Measurements are taken of alpha, beta, gas and flow rates.

The second circle is made up of eight monitoring stations on the perimeter of the plant site which help to determine if an off-normal event may have detrimental effects off site. Measurements are taken of the radioactivity of airborne elements and of ambient radiation.

The third circle is made up of five monitoring stations located in neighboring villages 1 to 5 kilometers from the site. Gas activity and radiation are measured in real time.

Drainage Systems

All water passing through the site is monitored before release to the surrounding environment by five COBENADE beta monitors. The flow rate is continuously measured, and complementary chemical analysis of water are also performed.

Meteorological Parameters

A meteorological station is installed on site. It will help to determine transfers of radioactive gaseous effluent releases to the atmosphere during both normal and accidental operating conditions.

Central Control Room

Monitoring data are centralized to provide an immediate overall view of radiological conditions at the plant and in the environment. Each monitoring station can automatically develop and display the data locally, and send it to the central environmental control station. The central control station's multi-tasking computer receives data in real-time from the various monitoring stations, immediately displays them on color monitors, prints out data reports and archives data files.

Delayed Monitoring

Delayed monitoring involves the taking of representative samples from the environment following a regular and periodic program, and analyzing them in the Environmental Laboratory operated by the Radiation Protection Department. The environmental monitoring program enables detailed dose calculations to be established; it involves some 17,000 samples a year taken from the three pathways for radionuclide migration to the food or biological chain -- atmospheric, hydrogeological, and marine -- and around 50,000 analyses.

Atmospheric Pathways

Continuous real-time monitoring is supplemented by measurements taken on filters and activated-carbon traps at each plant outlet and at the monitoring stations on the site boundary and in the outlying areas. Monitoring includes potential fall-out in rainwater, vegetation, crops, milk and meats.

Hydrological Pathways

The 32 springs and streams originating near the plant are monitored and analyzed. Through a site network of 220 piezometers, the water table can be closely monitored. The district's drinking water is also regularly analyzed.

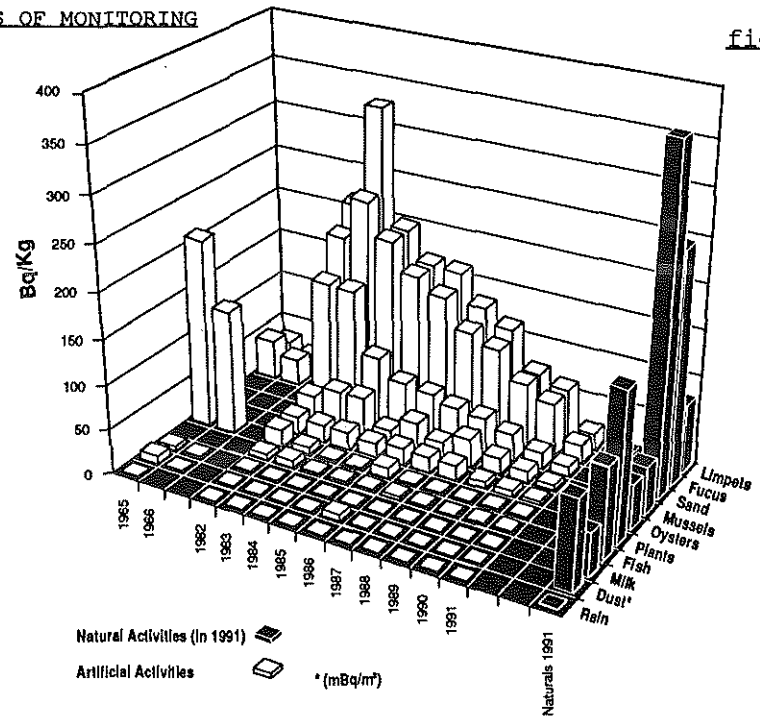
Marine Pathways

A 5 kilometer long submerged pipeline carries liquid releases from the plant out to the sea. The pipeline is regularly inspected. Two hundred kilometers of coastline, from Granville to Le Havre, are sampled, including water, sand, sediment, crustaceans, shellfish and plant-life which act as filters for radionuclides released into the sea. Deep sea sampling includes water, sand, sediments and fish. Sampling is performed by the French Navy, while the Marine Radioecology Laboratory of the French Atomic Energy Commission (CEA) studies marine dispersion of radionuclides in the English Channel and the North Sea.

The La Hague monitoring program, which is approved by the Ministry of Health, is a source of valuable data. Figure 1 shows environmental monitoring data for the last ten years, along with measurements taken in 1965 and 1966, the reference years for natural site conditions prior to the start-up of the plant.

4.RESULTS OF MONITORING

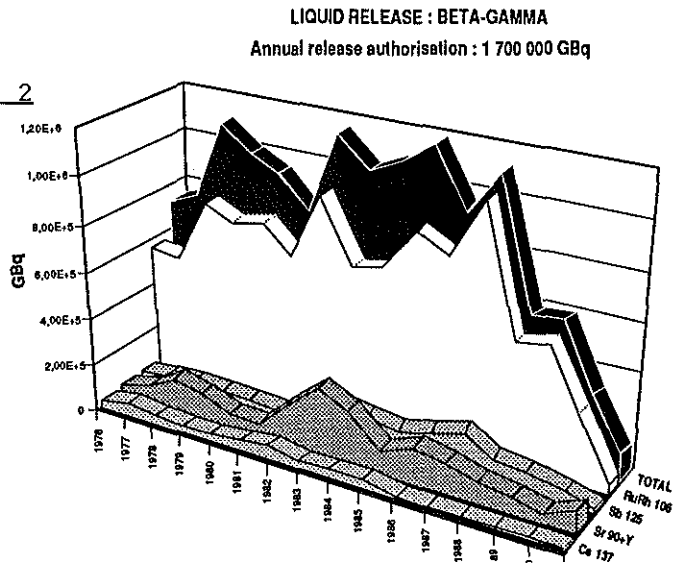
figure 1



The highest activity levels are also the oldest (at the back of the table), and relate to marine monitoring. Activity levels have decreased over the last ten years, while the quantities of reprocessed fuel continued to grow. This decline in activity reflects lower releases of activity to the sea (figure 2).

Figure 1 also shows that, for the last few years, man-made radioactivity has remained at the same level as natural radioactivity (potassium 40 and beryllium 7) for limpets, and even less for other samples.

figure 2



CONTROL AND DISCHARGE OF RADIOACTIVE LIQUID EFFLUENTS FROM THE *MARCOULE* COMPLEX

*F. LEVY, *A.CLECH, **J.M. GIORDANI, **J.P. MISTRAL
COGEMA/MARCOULE - BP 170 - 30206 BAGNOLS SUR CEZE
* Service de Protection contre les Rayonnements (SPR)
** Section de Traitement des Effluents Liquides (STEL)

Summary

With 34 years of experience in the field of nuclear industry, Marcoule site rallies differents activities. Even though reprocessing plant belong to COGEMA, the CEA is in charge of R & D. All the units produce radioactive wastes, wich are released in the Rhône River after a specific treatment in respect of national rules. These treatments are set off by the STEL (the liquid treatment unit) exploited by COGEMA. The Health Physics Department (SPR) is envolved in the control procedure of theses discontinuous discharges. Continuous sample stations, allowing the effluent dispesion control are managed by the SPR.Results about the last decade will be exposed here. 1991, year of the new liquid treatment launching will be more detailed.

Introduction

After 34 years of work in the nuclear field the MARCOULE complex now comprises a number of different activities. These include : On the one hand, the COGEMA installations for production purposes, with units involved in the reprocessing of irradiated fuels (fuel dismantling, the UP1 chemical separation plant, vitrification, etc) and the Celestin reactors.On the other hand,the CEA installations which are essentially involved in research and development,with laboratories,prototypes,industrial pilot plants and the Phenix reactor.These installations produce liquid radioactive effluents in large volumes (20000 m^3 p.y.) but with low radioactivity (6500 TBq) wich need treatment before the current regulations permit them to be discharged into the environment. Such treatment is carried out by the Liquid Effluents Treatment Station (STEL) operated by COGEMA. We will deal with the following headings in this paper :

1 Regulations
2 STEL functioning scheme
3 Effluent discharge
4 Report on the last ten years of discharging
5 Dilution surveillance

1 Regulations

Two regulations of the 20th, May 1981 govern liquid and gaseous discharges from the Marcoule complex. For liquid effluents these orders lay down the maximum quantities which may be discharged into the River Rhône together with applicable conditions . The annual activity of radioactive liquid effluents discharged by the entire COGEMA installations at Marcoule must not exceed : 2500 TBq for tritium, 150 TBq for radioelements other than ^{90}Sr,^{137}Cs and tritium ,6 TBq for ^{137}Cs, 6 TBq for ^{90}Sr, 150 GBq for alpha emetting radioelements. All effluents must be filtered prior to discharge so as remove all particles of a diameter greater than 25 micrometres. Discharges are spaced out over time so that in any one month they do not exceed one sixth of the corresponding annual limit. The flow of the Rhône must be between 400 and 4000 m^3/second. The activity concentration increase calculated after complete dilution in the Rhône must not be more than (in mean daily values) : 8 Bq/l for the aggregate of beta radioelements other than tritium, 100 Bq/l for tritium, 1 Bq/l for ^{137}Cs and ^{90}Sr, 40 mBq/l for the alphas.

2 Treatment in the STEL

The STEL was built in 1958 and has been developed since to deal with for the changing activities of the complex. In 1980 a decision was taken to renovate the installation and this resulted in the "new STEL" using a new process -evaporation- housed in a facility called EVA which comprises two evaporation lines of 8.5 tons per hour capacity. The EVA was started up on

the 3rd, April, 1990 and forms an integral part of the STEL process, wich consists of :
- installations for the collection (networks of buried drains), reception and storage of effluents (in reservoirs and tanks),
- a treatment plant comprising a coprecipitation unit and the new EVA facility,
- a bitumen embedding facility comprising a building known as the Embedding Preparation Building (BPE), a filtration line for the coprecipitation sludge and a bitumen embedding line equipped with a four-screw extruder,
- an installation for discharge into the Rhône consisting of a unit for filtration of decontaminated effluents, storage reservoirs, a sampling system for analytical control, predischarge filtration (25 micrometres) and a 5000 m^3 storage reservoir (normally empty) which acts as a buffer for occasions when it is not possible to discharge into the Rhône (low or high river levels),
- a centralized control room which enables the entire plant to be operated by remote control.

2-1 The chemical coprecipitation process for decontamination

Radioactive effluents which are not suitable for treatment by evaporation are subject to this chemical process which enables the major part of the radioelements in the effluents to be attached to a carrier precipitate. Three specific radioelements undergo treatment by successive additions of reagents in a basic medium : ruthenium (the most difficult to decontaminate owing to its presence in complex anionic, cationic an nonionic forms), strontium and caesium. This treatment also shows good results in the decontamination of cerium, zirconium, niobium, cobalt and alpha emitters. The precipitates thus rendered insoluble are flocculated and decanted at the end of the process. After checking the efficiency of the process, the remaining liquid is transferred to the discharge reservoirs via a first filtration line (50 micrometres), and the concentrated precipitates are taken to the bitumen embedding facility. The process currently in use gives decontamination factors of around 100 for beta (apart from tritium) and 400 for alpha (apart from uranium).

2-2 The evaporation process for decontamination

This is a semi-continuous evaporation process carried out at atmospheric pressure. High and medium activity effluents are treated in cycles whose length depends on the initial salts content (essentially sodium nitrate) so as to arrive at 400 g l^{-1} concentrates in the evaporator at the end of the cycle. The resulting distillates are led into a reservoir for neutralization and then go into the discharge line. Every 700 operating hours rinsing is done by successively boiling 3N soda, 3N nitric acid and water to remove accumulated deposits. Concentrates are transferred to the embedding preparation facility. The global decontamination factor for the installation is $5x10^4$.

2-3 STEL - overall performance

We can report that the environmental effects of discharging effluents from the new STEL, including the EVA facility, have been reduced by use of the coprecipitation process for part of those effluents which are not suitable for the evaporation process. Since 1991 discharges into the Rhône have thus reduced by a factor of 2 compared to previous years. The global decontamination factor for beta, apart from tritium, reached 267 for 1991. The alpha decontamination factor reached about 1000 .

3 Effluent discharge

The Radiation Protection Service guarantees that regulation requirements are complied with. They carry out checks which mainly consist of :

3-1 Checking the activity of discharges by analysis of samples taken from each reservoir before discharge takes place. The alpha and gamma spectra, the ^{90}Sr and ^{3}H contents, the pH (5,5 - 8,5) and suspended material (<100 mg/l) are all determined by analysis of the sample. An ionic analysis and TBP measurement are done monthly from an aliquot sample.

3-2 Making sure that activity discharged into the Rhône does not exceed the river's activity level above the permitted limits and giving authorization and setting conditions for the discharges. Discharge is done through a pipe laid across the bed of the river. The pipe has 13 outlet nozzles. Continuous checking of irradiation measurements and gamma spectrometry is done at the discharge pipe. Studies and modelling have enabled us to establish the optimal relationship between the velocity of the discharge (VD) and the river velocity (VR). This ratio VD/VR is 25, with acceptable upper and lower limits of 35 and 12. Using the forecast flow of the Rhône, the rate of discharge can be calculated so that the form and direction of the jets are optimal, meaning that they disturb neigher the river surface nor the river bed and that dilution is complete by the time the river enters the feed canal to the Caderousse hydroelectric station. Once the discharge flow rate has been set we must consider the maximum mean daily activity permitted to enter the river. If necessary we add a suitable quantity of plain water into the effluent so as to maintain the correct VD/VR ratio. With these parameters set, the Radiation Protection Service gives the STEL an authorization for the discharge.

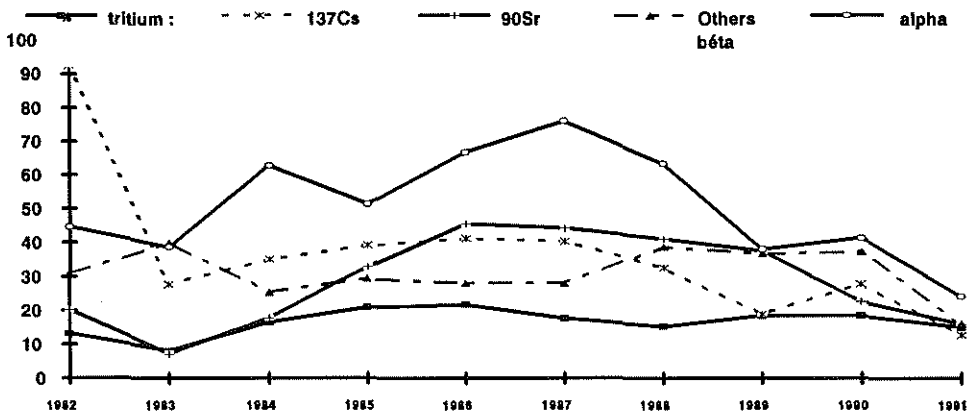
3-3 Record-keeping

The activity of the effluent discharged and all information in connection with the discharges (starting and stopping times, volumes, flows, etc) are filed and, according to regulations, reported to the SCPRI (the French organization for surveillance of ionizing radiation)

4 Results of the last ten years

Confining ourselves to the different categories of radioelements specified in the discharge authorization (see figure 1) we can report that :

Figure 1: LIQUID EFFLUENTS IN 1991:PERCENTAGE OF DISCHARGE AUTHORIZATIONS
Tritium:2500TBq--137Cs:6TBq --90Sr:6TBq-- Others beta:150TBq --Alpha:150GBq



4-1 Alpha emitters : having reached a maximum of 114 GBq for the year 1987, the activity discharged has decreased regularly since then. This applies particularly to 1991 (36 GBq discharged) owing to the EVA coming into operation. This activity represents 24% of the authorization.

4-2 Beta emitters apart from Cs and Sr : discharges were fairly constant from 1982 to 1990. The amount discharged in 1991 was greatly reduced owing, again, to the EVA. The discharge of 24 TBq represents 16% of the authorization.

4-3 Sr and Cs : for these elements the progression has been fairly parallel with a large drop between 1982 and 1983, followed by regular increases up

to 1986 and reductions again up to 1991,owing,again to the EVA. ^{137}Cs : 12,7% (0,76 TBq) and ^{90}Sr : 16% (0,96 TBq).

4-4 The activity relating to tritium has been constant and amounts to about 400 TBq which is 15% of the authorization.

If we go a bit further into detail concerning the compositions of the various alpha emitters we find that the proportion of each of the emitters natural U, $^{239+240}Pu$, $^{238}Pu+^{241}Am$ and $^{242+244}Cm$ showed relatively little change over the years . We note, however, that the proportion of $^{239+240}Pu$ tended to increase regularly from 20% of the alpha emitters in 1982 to 35% in 1991, while the other elements reduced correspondingly.

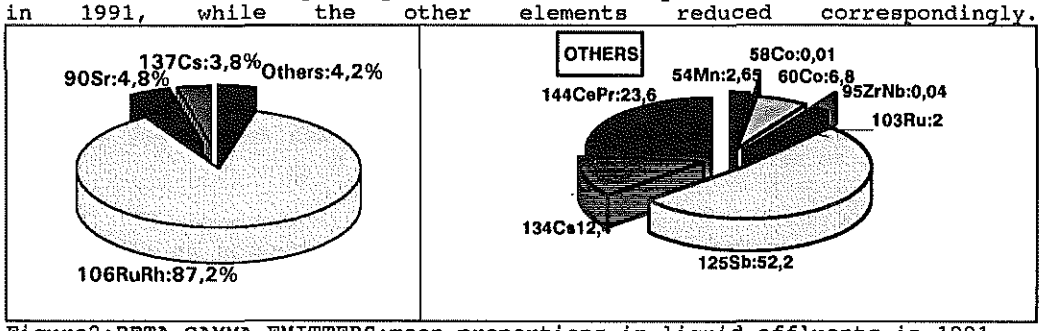
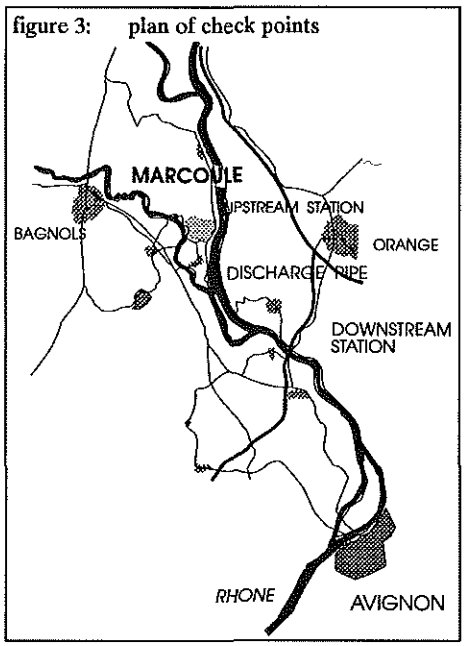


Figure2:BETA-GAMMA EMITTERS:mean proportions in liquid effluents in 1991

The major part of the discharge of beta and gamma emitters consists of $^{106}RuRh$, amounting to 17,4 TBq in 1991, in which year it accounted for 87,2% of the beta and gamma discharges. In the previous ten years it, fluctuated between 76% in 1982 (discharge of 36,6 TBq) and 92% in 1983 (discharge of 53,3 TBq). Beta activity apart from ^{3}H reflects essentially the progression of $^{106}RuRh$ discharges. The other significant beta and gamma emitters present are, in diminishing order, ^{90}Sr, ^{137}Cs, ^{125}Sb, $^{144}CePr$ and ^{134}Cs (see figure 2).



5 Surveillance of effluent dilution in the Rhône

The dilution of the effluents in the Rhône and the rise of activity in the river water are checked by continuous sampling, the samples being analysed in the laboratory. This continuous checking is done at 2 stations on the river : one upstream of the discharge pipe and the other at Roquemaure at kilometre marker 222 which is 11,5 km downstream of the discharge and below the Caderousse dam.

5-1 Checking the river water is done in two ways :

1) analysis at mid-discharge
2) weekly and monthly analysis of aliquot samples.

5-1-1 Sampling at mid-discharge consists of taking a water sample half way through the time taken for each discharge. The time taken to arrive at the sampling station is calculated from the speed of the river during the discharge. Working from the starting and stopping times of the discharge the sample can be collected at the moment which corresponds to the mid-point in the discharge. For this purpose a battery of water collectors operates all the time. From each sample we determine the global beta,total alpha and tritium to ensure that the regulations are complied with .

5-1-2 Making up of mean weekly an monthly samples is done using another set of water collectors. This is done at both the upstream (RI) and downstream (RIII) stations. The water first passes through a decanting tank, in which the sludge is analysed monthly, and is then collected in drums. Each week we determine the total beta (soluble - insoluble), uranium, tritium and potassium. Each month we also determine the ^{90}Sr (by extraction of yttrium an measurement by the Cenrenkov effect) and the gamma spectrum from a sample of 60 litres.

5-2 Results of surveillance

Tritium : the chosen detection limit for these analyses is 22 Bq/l and the mean activity measured over one year remains below 25 Bq/l.

Global beta activity : this is measured by comparison with a standard sample of ^{90}Sr using a detection limit of 0,24 Bq/l. The annual means taken from the weekly measurements are shown in the figure 4. The maximum values were those of 1989 and 1990, years when the Rhône was at its lowest levels(1028 m^3/s), when 0,8 Bq/l was recorded.

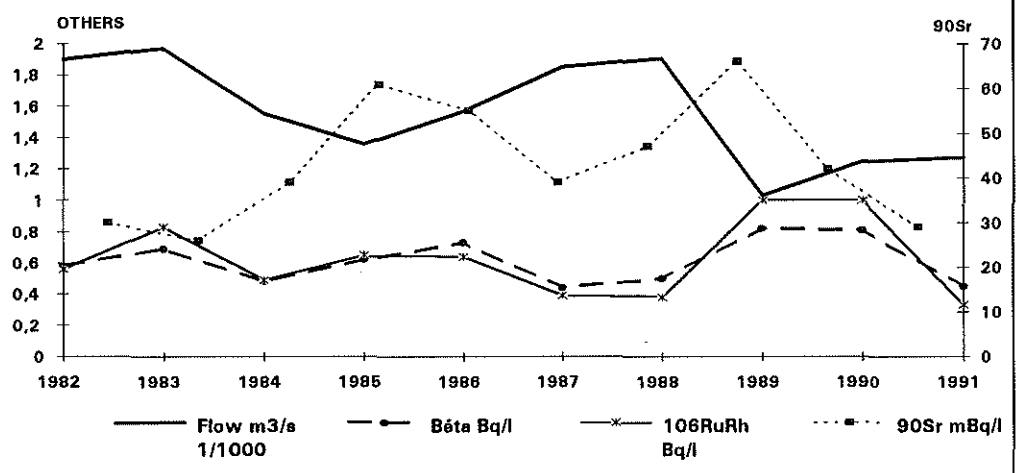
It is more instructive to consider the radioelements separately. The most commonly detected in the water are $^{106}RuRh$, ^{90}Sr and sometimes ^{125}Sb:

106RuRh, the major constituent of our effluent shows the largest proportion in the water as well. It fluctuates in the water between 0,33 and 1 Bq/l. Figure 4 shows a graph of mean annual activity of $^{106}RuRh$ over the last ten years, expressed in Bq/l. This activity depends on the quantity discharged and also on the flow of the river. The activity in the water is in inverse proportion to the flow of the river, since, assuming constant discharge, increased flow results in an increased dilution.

^{90}Sr : Representing on average 4 to 5% of our discharges, it fluctuates between 26 mBq/l and 66 mBq/l

. On the other hand, the ^{137}Cs, of which about the same average quantity is discharged, is hardly ever detected in the river water (<15 mBq/l) by our routine measurements. This is explained by its affinity for clay particles suspended in the water.

figure4 PROGRESSION OF THE RHONE WATER RADIOACTIVITY



6 Conclusion

In addition to the above observations we would point out that our role consists not only of surveillance of discharges and river water. The various constituents of the aquatic ecosystem are sampled throughout the year and analysed either for our own purposes or in conjunction with the CEA/IPSN. The results of these studies form a part of Mr FOULQUIER's paper.

RADIOACTIVE EMISSIONS WITH THE EXHAUST AIR OF THE *KARLSRUHE NUCLEAR RESEARCH CENTER* AND THEIR IMPACT ON THE ENVIRONMENT

A. Wicke, M. Winter,
Kernforschungszentrum Karlsruhe GmbH,
Hauptabteilung Sicherheit/Umweltschutz,
D-7500 Karlsruhe, Germany

Abstract

Radioactive substances are released with the exhaust air of the nuclear research facilities of the Karlsruhe Nuclear Research Center (KfK). In observance of the dose limits to the public prescribed by law, the so-called "Effluent Plan" of KfK prescribes levels of maximum permissible releases of radioactive substances with exhaust air for each individual emitter at KfK. These releases are monitored and recorded by specific measurement and collection systems.

To estimate the radiation exposure of the public, dose levels are determined at the most adverse points of impact, in accordance with the "General Administrative Regulations of Sec. 45 of the German Radiation Protection Ordinance," by superimposing the dose contributions made by individual emitters and calculating them for all exposure pathways and organs. This is done on the basis of the monthly meteorological dispersion conditions as determined on site and the monthly releases of emitters. As a consequence of the decommissioning of research reactors over the past few years and the discontinuation of work on reprocessing spent nuclear fuels, radiation exposure in the environment has clearly decreased. At present, the effective comitted dose equivalent summed up over all exposure pathways at the most adverse point of impact is around 1 µSv per annual activity release.

1. The Nuclear Research Center

Major emitters in the sixties, the early phase of KfK, were the Institute for Radiochemistry, the Hot Cells Facility, the Institute for Hot Chemistry, and the FR2 and MZFR research reactors. The reprocessing plant for spent nuclear fuels (WAK) was commissioned in 1971, while KNK II, the fast breeder research reactor, was started up in 1977. In the fields of decontamination and waste treatment a number of facilities were built for evaporating process solutions and for conditioning low and medium-level radioactive wastes.

Other sources of emissions are the cyclotron, which is used to produce shortlived radioactive tracers for medical application, the European Institute for Transuranium Elements, and the Laboratory for Aerosol Physics and Filter Technology.

Radioactive emissions arising when spent nuclear fuels and related radwaste are being handled are characterized largely by releases of radioactive fission products and fission product mixtures. One exception to this rule is the cyclotron, whose releases are dominated by shortlived activation gases.

The research reactors have been decommissioned step by step over the past few years, and reprocessing work has been phased out in 1991. This has caused radioactive releases to drop markedly. Major sources of emissions continue to be the facilities for decontamination and for the treatment of radioactive waste. In the

field of fusion research, releases by the Tritium Laboratory will have to be taken into account in the future.

Radioactive substances are released through distributed exhaust air vents and stacks with heights between 5 m and nearly 100 m. Compared to industrial plants, such as nuclear power plants, the releases from these facilities exhibit greater variability both in terms of activity composition and release rates.

2. Effluent Plan

Section 45 of the German Radiation Protection Ordinance [1] prescribes that releases of radioactive substances associated with the operation of nuclear facilities be limited in such a way as not to exceed certain dose limits to the public, i. e. 0,3 mSv/y effective dose equivalent. For this purpose, maximum permissible release levels with the exhaust air are defined individually for each emitter of KfK, which levels are based both on requirements of operation and on the need to minimize emissions.

The maximum permissible releases are listed in the so-called "Effluent Plan" broken down by emitters and nuclides or nuclide groups. This plan is updated once a year and submitted to the supervisory authority for approval. The following groups of nuclides and individual nuclides are distinguished in it:

- Aerosols with alpha-activity,
- aerosols with beta-activity,
- radioactive noble gases and activation gases,
- radioactive iodine,
- tritium,
- radioactive carbon (C-14).

Tracer nuclides and the compositions of the characteristic nuclide mixture have been defined specifically for each plant on a conservative basis. In these definitions, both knowledge of specific modes of operation and experimental programs run by the emitters, data about their radioactive inventories and their potential releases as well as analytical results of the releases measured were taken into account.

The Effluent Plan can be approved only if a prognostic calculation based on the model contained in the "General Administrative Regulations of Sec. 45, Radiation Protection Ordinance" [2] demonstrates that the limits under Sec. 45, Radiation Protection Ordinance, are observed. Details of the methods of dose calculation can be taken from chapter 4.

3. Monitoring and Accountancy

In order to protect the environment and to observe the maximum permissible release levels set forth in the Effluent Plan, radioactive emissions are monitored continuously. For this purpose, measurement and collection systems with representative sampling facilities have been installed in the bypasses of the ex-vent air stacks. Direct reading equipment (monitors) for radioactive aerosols, iodine, tritium, and noble gases directly indicate increased releases. Special collectors are used for accountancy of the activities released, except for the radioactive noble gases:

radioactive aerosols:	collection on fiberglass filters,
radioactive iodine:	adsorption to activated carbon,
tritium:	collection in molecular sieves (3 Å),
radioactive carbon (C-14):	collection in molecular sieves (10 Å).

The collection media are replaced at weekly and monthly intervals, respectively, and the samples are counted in a laboratory [3]. In 1991, more than 4000 such samples were analyzed. Releases may vary strongly both in terms of time and in nuclide composition and source strength. The results are recorded, arranged by time and by emitters and nuclide groups, and communicated to the respective institutes and to the licensing authority once a week.

Emissions have clearly decreased over the past ten years. Figure 1, by way of example, shows the development of tritium releases from all emitters over the past 22 years.

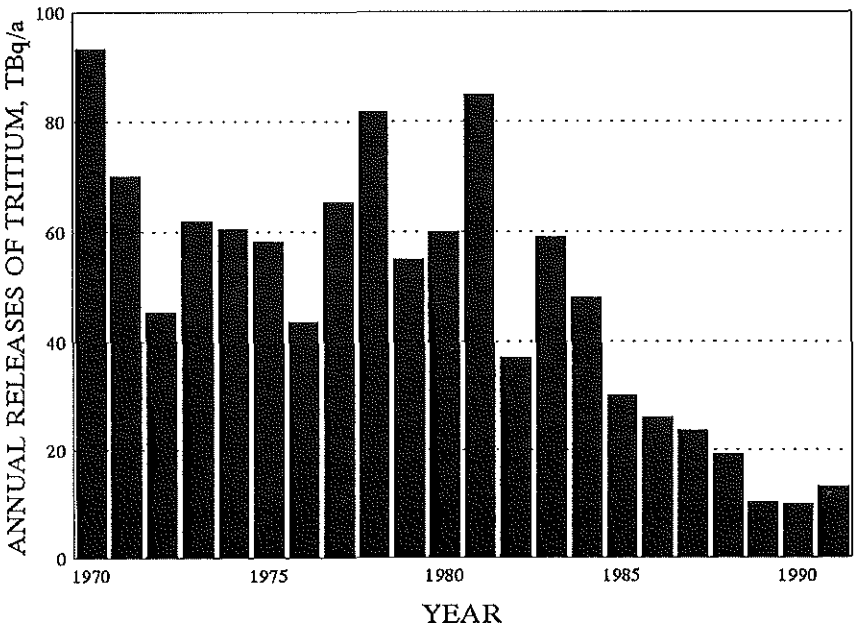


Figure 1: Tritium releases of KfK with the exhaust air.

In 1991, these activities were released by KfK with the ex-vent air:

Aerosols with alpha-activity:	1.9×10^{6} Bq,
aerosols with beta/gamma-activity:	2.3×10^{8} Bq,
radioactive noble gases:	9.9×10^{12} Bq,
I-129:	3.8×10^{7} Bq,
I-131:	4.9×10^{6} Bq,
tritium:	1.3×10^{13} Bq,
radioactive carbon (C-14):	3.9×10^{10} Bq.

4. Dose Calculation

Doses are calculated on the basis of the monthly releases of all emitters as measured. They are determined in accordance with the model contained in the "General Administrative Regulations of Sec. 45 of the German Radiation Protection Ordinance" [2]. The maximum possible doses for the most adverse point of impact in the environment of KfK, with all possible exposure pathways taken into account, are relevant with respect to the observance of limits under the Radiation Protection Ordinance. The calculation performed in accordance with the regulations listed above is conservative in the overall outcome. It is based, inter alia, on the assumption of specific dietary habits of a reference person, assuming that that person will consume only food whose original agricultural products were grown at the place of maximum contamination. In addition, the nuclides are assumed to have accumulated in the soil over fifty years.

It is not taken into account in this calculation whether there is really a possibility of somebody staying permanently at the most adverse points of impact, and whether the food items considered are actually produced there.

As provided for in the General Administrative Regulations, the dose factors required to calculate partial body doses and the effective dose resulting from inhalation, ingestion, and external exposure are taken from the "Bundesanzeiger" [4]. In order to enable relevant classes to be chosen for retention in the lung and for solubility in combination with the ingestion of radioactive aerosols, the chemical forms dominating in, or typical of, the different emitters are used as a basis on which to determine aerosol releases. Where they are unknown, conservative assumptions are made. The daughter nuclides are also taken into account in calculating dose levels.

The meterorological data needed for dispersion calculations are measured at the 200 m high meteorological tower on the premises of KfK. Wind direction, wind speed, and dispersion category are averaged every 30 minutes. Their frequency distributions are combined in the dispersion statistics. The compass rose is subdivided into twelve sectors of 30° each. The wind speed and wind direction at 60 m altitude is used as a basis in calculations.

Unlike the provisions in the General Administrative Regulations, an azimutal isodistribution not of the activity concentration, but of the frequencies of wind directions within a sector is assumed in the dispersion calculation. This is factually more correct and avoids ramps at the boundaries of sectors.

In determining the radioactive substances plated out as a result of dry deposition, the deposition rates indicated in the General Administrative Regulations for elemental iodine, organically bound iodine, and aerosols are taken into account. Calculations of depositions due to precipitations are based on the four-parameter weather statistics. For ingestion, the activity deposited on plants is taken into account only in the summer half of the year.

The dose contributions resulting from beta submersion, inhalation, ingestion, and gamma radiation over contaminated ground are determined for all field points of impact by superposition of the effects of all individual emitters. This is done by means of the ISOLA computer code [5] in combination with the EFFDOS computer code [6] for all organs and the effective dose equivalent.

In calculating the gamma submersion dose, the gamma dose as the sum total of the dose contributions made by the activity distributed in the region must be calculated for each field point. For this purpose, the WOLGA computer code [7] is

used. It furnishes the gamma dose for any field point in the environment of one or more emitters as the sum total of the dose contributions made by the activity in that region.

Under the boundary conditions described above, the partial body doses and the effective dose equivalent to infants and adults in the environment were calculated. For the most adverse points of impact outside the KfK premises, arranged by the exposure pathways to be taken into account, these maximum contributions to the effective dose equivalent result for 1991:

Exposure pathway	Infants	Adults
Inhalation	0.05 µSv	0.07 µSv
Ingestion	1.0 µSv	0.8 µSv
Gamma submersion	0.2 µSv	0.2 µSv
Gamma radiation over contaminated ground	0.05 µSv	0.04 µSv
Sum total of all exposure pathways	1.3 µSv	1.1 µSv

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ASSESSMENT OF RADIATION DOSES TO THE PUBLIC RESULTING FROM GASEOUS DISCHARGES FROM NUCLEAR SITES IN THE UNITED KINGDOM

M G SEGAL

Ministry of Agriculture, Fisheries and Food, Ergon House, c/o Nobel House, 17 Smith Square, London SW1P 3JR, United Kingdom

Summary

The Ministry of Agriculture, Fisheries and Food (MAFF) is one of the departments responsible for authorising the discharges of radioactive waste from major nuclear sites in the UK. Before any authorisation is granted, the potential dose to members of the public is assessed to ensure that dose constraints are not breached. The assessment of critical group doses due to atmospheric discharges is based on realistically conservative assumptions, which ensure that the results are overestimates rather underestimates of dose. Environmental monitoring demonstrates that true doses are considerably lower, confirming that there is a significant safety factor built into these assessments.

1. Introduction

Inspectors from MAFF have responsibilities for some 35 nuclear sites in England and Wales, including nuclear power stations, reprocessing facilities, research establishments and isotope production and processing plants. These sites may discharge radioactive wastes only under the terms of authorisations issued by the appropriate government departments, of which MAFF is one, and this paper describes aspects of the methodology used in the assessment of critical group doses due to discharges to atmosphere prior to the issuing of authorisations.

2. Authorisation of Discharges

All recent authorisations include numerical limits on the more important radionuclides, as well as a requirement to use "best practicable means" to limit discharges. In general, limits are placed on the annual totals of discharges, but shorter-term limits may also be imposed if necessary. Discharge authorisations also impose conditions on the operators, such as a requirement to monitor the discharges and the environment [1].

3. Dose Limits and Constraints

The principal dose limit for members of the public recommended by the ICRP remains 1 mSv [2]. However, the effective constraint in ICRP terms is the Government's target for individual authorisations that the committed effective dose equivalent to the critical group should be no greater than 500 μSv per annum [3]. (This constraint may be reduced in the future [4].) Before any authorisation is granted, the potential dose to the public is assessed, to ensure that the above target is not breached. The assessment of radiation doses that the public might receive as a result of discharges is inevitably an inexact process, and so conservative methods are used to ensure that the assessed dose is over- rather than under-estimated.

4. Assessment Methodology

4.1 Basic Principles

The basic approach used by MAFF is to determine which members of the public in the vicinity of each site are likely to receive the highest dose from its discharges, the critical group; if this group receives a tolerable dose then all other members of the public are adequately protected. All discharge pathways are assessed to find the overall critical group for the site [1]. For those exposed to the effects of gaseous discharges, the total dose must be assessed due to external γ-shine from inert gases, and

to internal doses from inhalation and from ingestion of both terrestrial foodstuffs contaminated by discharges to atmosphere and aquatic foodstuffs contaminated as a result of discharges of liquid effluent. In all cases, potential doses are assessed for discharges at the limits proposed for the authorisations. MAFF uses computer codes to model the atmospheric dispersion of effluents from the site and the movement of nuclides through the food-chain [5]; the basic methodology has been described elsewhere [6]. These models, and the assumptions made in their application, follow the principles laid down in ICRP Publication 29, and are based on "realistically conservative assumptions" [7].

4.2 The Critical Group

Most UK nuclear sites are in remote locations, and so the population in these locations tends be sparse, and often comprises farmers or agricultural workers. Discharges to atmosphere are dispersed from the site according to meteorological conditions and distance. Thus the population likely to be most exposed is that living close to a site, and in the direction of the prevailing winds: the potential dose to local inhabitants is very dependent on their precise location. It is therefore not possible in most cases to find a homogeneous group [8]; if necessary, the critical group may be as small as a single individual, if that person may receive a significantly higher dose than the rest of the population under consideration, solely as a result of where they live, rather than extreme habits. It is also possible for an infant or child to live in the critical habitation, even if there are none living there at the time of the assessment, and so the potential dose to all age groups is assessed.

4.3 Consumption and Consumption Rates

In an agricultural community it is clearly possible that many of the foodstuffs consumed on a regular basis are produced locally. It is thus not unreasonably conservative to assume that all foods that are, or can be, produced at the location of interest are eaten by the critical group. So the contamination levels are calculated for locally-grown foods, including milk, either at the critical habitation or at the nearest site of commercial production. In general, the consumption rates used by MAFF for high-level consumers of all foods are based on the 97.5 percentile of the national statistics for consumers. For dose assessments, it is now considered excessively conservative to assume that critical group members consume all foods at this high rate. The diet is divided into 12 food groups, such as milk, leafy green vegetables, root vegetables, etc. Ministry statistics show that an adequate degree of conservatism is assured by assuming high-rate consumption of the two food groups that contribute most to the dose, and average consumption for the rest.

4.4 Inert Gases: Ar-41

Nuclear power stations in Britain are currently all gas cooled, and all release Ar-41, some in large quantities. The dose due to external γ-radiation by this nuclide can be significant. Current practice is to assume 90% occupancy, and a shielding factor of 0.5 for a stone or brick house, and 1 (ie no shielding) for a wooden one. (These figures are under review.) In some cases, the dose due to Ar-41 is much greater than that due to ingestion of other nuclides, and it is important to ensure that all pathways have been fully considered before the critical group is defined.

5. Results of Assessment Calculations

Many new authorisations have been issued in the last few years, and the revision of authorisations is underway for many other sites. Table 1 shows some examples of the results of the assessments of doses due to atmospheric discharges from various sites of different kinds. Results of the Ministry's environmental monitoring programme confirm that these assessments are significantly pessimistic, and the doses received by the public through the food-chain are in reality much lower [9]. Thus each assessment contains a significant built-in safety factor.

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Table 1. Critical group dose assessments for atmospheric discharges from some UK nuclear sites.

SITE	TYPE	MAJOR NUCLIDES	ANNUAL LIMIT TBq	ASSESSED DOSE µSv	TOTAL µSv[b]
Sizewell A	Steel-PV	Ar-41	3000	140	
Power Station[a]	Magnox	S-35	0.6	7	
		C-14	1.5	5	
		H-3	7.5	<1	152
Wylfa	Concrete-PV	C-14	2.4	41	
Power Station	Magnox	Ar-41	120	12	
		S-35	0.5	10	
		H-3	20	2	65
Heysham	AGR[c]	C-14	4	14	
Power Station[a]		Ar-41	400	9	
		S-35	0.5	4	
		I-131	0.01	3	30
Amersham[d]	Isotope	Rn-222	10	84	
	manufacture/	I-125	0.1	64	
	processing	I-131	0.05	24	
		Se-75	0.03	3	210

NOTES: (a) Limits quoted are currently proposed, not in force.
(b) Including minor components not listed individually here.
(c) Total for Heysham 1 and Heysham 2 stations.
(d) Discharges for this site are much lower than the limits.

EVALUATION OF THE RADIOLOGICAL IMPACT IN THE ENVIRONMENT AROUND *IPEN's* (*) FACILITIES

V.M.F. JACOMINO, A.M.P.L. GORDON, M.F. MADUAR
(*) Instituto de Pesquisas Energéticas e Nucleares.
Comissão Nacional de Energia Nuclear, São Paulo.
Cx. Postal 11.049 - CEP 05422-970.

Abstract

In order to control the discharges of radioactive material generated by the IPEN's facilities, an effluent monitoring program was established on a routine basis. This control is carried out by measuring the activity of the radionuclides present in the effluents (sourceterm) using gamma spectrometry and/or neutron activation analysis. The results obtained are then compared with the operational limits adopted, when a decision is made upon the discharge of the effluents. In this paper the data concerning the sourceterm from 1988 to 1991 as well as the effective dose in the critical group are presented. All the results are below 1/10 of the limits of dose recommended by the Radiological Protection Standards. These data together with those obtained from the environmental monitoring program prove that the radiological impact due to the discharges of radioactive effluents by the IPEN's facilities is negligible.

1. Introduction

The Environmental Monitoring Division of IPEN has established on a routine basis an environmental monitoring ([5]) program in order to determine the amount of radioactive material discharged to the environment as well as to detect a non planned release (above the pre-selected operational limits). In this report the results concerning the sourceterm released from 1988 to 1991 are presented. The effective dose in the critical group as evaluated by using the sourceterm and a generic model([3]) which describes the transfer of the radionuclides into the ecosystem. The analysis of these results showed that the radiological impact around IPEN's facilities is negligible.

2. The Nuclear and Radioactive Facilities Available at IPEN

The nuclear and radioactive facilities available at IPEN which contribute to the liquid and gaseous sourceterm are: the swimming pool research reactor IEA-R1, which operates at a nominal power of 2 MW, responsible for medical, engineering, industrial and research purposes and a Centre for the production of labelled compounds and radioisotopes such as I-131, P-32, Cr-51 used in nuclear medicine.

Other facilities such as the decontamination laboratory and a centre for development of the main steps of the nuclear fuel cycle, give rise only to liquid effluents.

3. Radioactive Effluents Monitoring System

The liquid effluents are kept in special tanks. These effluents are sampled (1 liter) and sent to the Environmental Mo

nitoring Division for the determination of the radioactive contents. The activity is measured by using a hyperpure germanium detector with 15% efficiency, coupled to a 4096 multi-channel analyser. The samples which contain uranium are analyzed by neutron activation analysis. The final results are corrected for the total volume available in the tank. The activity obtained is compared with the daily discharge limits adopted by the Brazilian Nuclear Energy Commission([1]) and a decision is made upon its discharge to the environmental. After the authorization has been given the effluents are directly discharged in the sewage system and flow to Pinheiros river. There are no evidence of aquatic life in Pinheiros river since it receives continuously a considerable amount of industrial and domestic water.For that reason the river water is not used for irrigation purposes, not used as drinking water supply for the population([5]). Only some factories located at the river bordeline pump the water for industrial machines refrigeration.

The gaseous effluents are continuously monitored by using a air sampler coupled to a flux meter. This system is conected off line in the stack. The airbone and gaseous effluents are collected through a filters system arranged after the pump. The filters are routinely measured by using the same equipment already described for the liquid effluents. The results are corrected for the air flux in the stack.

4. Sourceterm Data From 1988 to 1991

The liquid effluent activity discharged per year by IPEN's facilities is presented in table 1. From this table, were excluded, the radionuclides wich activity contributed with less than 1% per year and which were not considered critical.

Table 1: Liquid effluents activity discharged per year(1988-1991)

Radionuclides	Total activity released (Bq)			
	1988	1989	1990	1991
Na-24	$3,1.10^7$	$9,9.10^6$	$8,8.10^7$	$1,2.10^7$
Co-58	-	$1,3.10^6$	$5,7.10^6$	$1,3.10^6$
Co-60	$1,6.10^8$	$3,4.10^8$	$4,4.10^8$	$3,8.10^8$
Zn-65	$1,8.10^7$	$1,7.10^7$	$5,4.10^5$	-
Tc-99m	$4,1.10^5$	$2,6.10^6$	$1,1.10^7$	$1,3.10^6$
Ag-110m	$2,3.10^6$	$9,9.10^5$	-	-
I-131	-	$4,4.10^7$	$6,6.10^7$	$2,4.10^7$
Cs-134	$8,9.10^6$	$1,5.10^7$	$6,6.10^7$	$2,4.10^7$
Cs-137	$7,5.10^7$	$1,4.10^8$	$1,2.10^8$	$6,4.10^7$
U-nat	$1,5.10^8$	$2,3.10^9$	$8,0.10^5$	$9,7.10^7$
Total volume released (m^3)	2421	1242	1377	1088

The gaseous and airbone effluents activity discharged from 1988 to 1991 is presented in table 2. Just I-131 was detected which activity above the discharge limits adopted ([1]).

Table 2: Gaseous and airbone activity discharged per year (1988-1991)

Year	I-131 activity released (Bq)	
	Gaseous	Airbone
1988	$6,5.10^{10}$	$5,6.10^{6}$
1989	$9,3.10^{9}$	$8,2.10^{6}$
1990	$1,7.10^{10}$	$9,4.10^{6}$
1991	$1,1.10^{10}$	$5,2.10^{6}$

5. Effective dose in the critical group

As was pointed out he Pinheiros river water is not used for irrigation purposes or for drinking water consumption. Therefore, for the liquid effluents, the only contamination pathway to be considered is the dispersion and sedimentation of the radionuclides, giving rise to the gamma external irradiation. The critical group is formed by those individuals of the public that work at the river bank near the discharge point([4]).

For gaseous and airbone effluents, the critical group is formed by the people living 3000 m away from the discharge point in the north-west diffusion section. The main exposure pathway is the gamma external irradiation due to the deposition of radionuclides on the ground([2]).

The anual effective dose in both cases was calculated for each radionuclide listed in table 1 and 2. These estimates have been carried out by using the IAEA transfer model([3]), and by applying the proper dosimetrie factors([3,5]).

The effective doses in both critical groups are showed in table 3 and 4, for liquid and gaseous effluents, respectively.

Table 3: Effective dose in the critical group due to liquid effluents discharge.

Year	Effective dose (mSv/year)
1988	$2,7.10^{-5}$
1989	$5,6.10^{-5}$
1990	$7,0.10^{-5}$
1991	$5,6\text{-}10^{-5}$

Table 4: Effective dose in the critical group due to gaseous and airbone effluents discharge.

Year	**Effective dose (mSv/year)**
1988	$1,1.10^{-3}$
1989	$1,6.10^{-4}$
1990	$2,9.10^{-4}$
1991	$1,9.10^{-4}$

6. Conclusion

During the last 4 years no considerable changes were observed in the operation conditions of the IPEN facilities, as showed the results of the sourceterm in table 1 and 2.

The radionuclides which more contributed for the effective dose due to the liquid effluents discharge were Co-60 (79,9%), Cs-137 (18,2%) and Cs-134 (1,4%). These radionuclides come from the treatment process of the reactor pool water.

As was already point out, I-131 was the only radionuclide detected in the gaseous and airbone effluents arising from the radioisotope and labelled compounds production.

The effective dose in the critical group has been always below 1/10 of the dose limit recommended by the Radiation Protection Standards([1]). Therefore the environmental impact due to the normal operation of the IPEN's facilities is negligible.

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THE NEW STACK EFFLUENT MONITORING SYSTEM IN *MÜHLEBERG* NPP

M. Haller, B. Blaser and C.-D. Schegk

1. Introduction

The emission control system of the stack effluent had to be newly designed after the release of radioactive aerosols in September 1986 caused by a filter defect. Lay-out, solving of problems and operational experience are topics of this article.

2. Requirements

The minimum requirements by the licensing authorities were:
- measurement of all conveyable aerosol sizes
- representative sampling
- registration and alarm in the control room
- diverse aerosol measurement, independent of a sampling system
- iodine monitoring

Particles of 0.8 mm in diameter, based on a density of 2000 kg/m^3, can be conveyed in the stack. Additionaly to the aerosole measurement, therefore, a particle measurement becomes necessary.

Representative sampling has to be provided by isokinetic extraction in the stack for variable effluent flows (operation of ventilation). Registration and alarm in the control room must be adjusted to the local readings and alarms. Diverse aerosol measurement, independent of a sampling system, means twofold realization of sampling, piping, and measuring systems.

As the former system lacked one iodine monitor, this also had to be upgraded.

The existing aerosol and noble gas monitoring devices were kept.

3. Lay-out of the New System

To fullfill the requirement of a diverse aerosol measurement independent of a sampling system, essential parts of the monitoring system had to be built twice. The original system had its extraction probes at +120 m with a delivery line to the measuring room at the foot of the stack. This arrangement was kept in the new design as partial system 0 m. An existing elevation system outside the stack and an overpass within at +90 m led the erection of the second partial system at +90 m.

4. Partial Systems

The air sampling is made through to spider - like grids of 10 sampling probes each. All probes are located at the same stack level and alternatingly set on two circles according to the colloidal line method. The pipes are gathered in a tube unifier, and the main tube is led down along the stack wall (Fig. 1). The flow velocity in the train of pipes of > 20 m/s is achieved by use of side channel pumps and results in a total flow through each leg of 300 m^3/h.

One presupposition for the samples' representativity is their isokinetic extraction from the effluent. It occurs as soon as the flow velocity in a probe is equal to the flow velocity at the sampling locus in the stack.

To provide for isokinetic conditions at all times, the sampling flow is automatically adjusted to the effluent. The regulation is achieved by comparing the stack flow velocity (two vane anemometers at +90 m) to that inside the delivery line (one vane anemometer), and controlled adjustment of a bypass flap at the suction side of the pump. The delivery line is isolated and heated where it passes through the cold stack foot to avoid condensation within the tube. The sampling probes of the individual measuring devices are isokinetic to the average velocity in the main tube and are not regulated. The local positioning of the probes conforms to DIN 25423.

5. Particle Measurement

The fact that large particles can be emited makes their registration mandatory. This is achieved with the ABB large particle detector (ABB-GAD). This apparatus was especially designed for the detection of large, radioactive particles. It consists of two partial devices, a classifier and a detector. The air sampled in the stack is led to the classifier. Consider particles of an aerosol having a size frequency distribution, q_A (dp). In the classifier they are sorted according to their size. The aerosol's size distribution is split into a fine fraction, q_F (dp), an a coarse fraction, q_G (dp). The correlation between the two functions is the grade efficiency curve of the classifier:

$$T(dp) = 1 - f \frac{q_F(dp)}{q_A(dp)} \cdot \frac{g \cdot q_G(dp)}{g \cdot q_G(dp) \cdot f q_F(dp)}$$

with f an g being the relative mass fractions of fine and coarse matter respectively of the total particle mass. The 50% cut rate is set at an aerodynamic diameter of 10 µm.

After being classified the fine particles leave the separator as an aerosol, where as the large particles sediment upon a filter. This filter is monitored by a y-detector and can be taken out for laboratory analysis. The aerosol flow is led through a grab filter after the classifier to retain the suspended particulate matter. The particule-free air passes through a charcoal bed for iodine separation before being blown back to the stack.

6. Instrumentation of Sampling Leg +90 m

Because of its special location in the stack at +90 m, this leg was only equipped with a minimum of measuring devices. Large particles are monitored with an ABB-GAD, and fine aerosol with a ribbon filter. Inventory sampling of large and fine particles is done with the grab filters of the ABB-GAD as well as iodine collection with its charcoal filter.

7. Instrumentation of Sampling Leg 0 m

This leg comprises the total instrumentation. Additional to the ABB-GAD one finds a ribbon step filter, the iodine monitor, and noble gas monitoring.

8. Additinal Systems

- Control Electronics
- Power Supply
- Lightning Protection
- Protection against Steam Invasion
- Emergency Measurement

9. Commissioning Tests

During the transport of the samples from the nozzles in the strack to the measuring devices, particles are inevitably lost on the walls of the piping. The authorities expected losses not to exceed 50% for all conveyable particle sizes. For this reason passage rate test had to be conducted for small and large particles. The materials used were stell beads (Grannacciaio) as large particles with 70% mass consisting of particles between 0.18 and 0.35 mm, and sodium sulfate as fine particles having a particle size spectrum from 0-12 μ-m.

10. Experimental Set-up

The measurements with large particles were conducted using a pneumatic dispersion machine situated right below an extraction probe. Particles sucked into the machine from a funnel were injected via an elbow into the nozzle end, from where they were taken in by the system and passed on.

The set-up for fine particles consisted of a tube with funnel of approximately 1 m length connected to a probe, and an ultrasonic mist generator feeding droplets into this antechamber. The sojourn time of the droplets was around 1 s, sufficient to allow their drying. A small extraction probe could be installed in the connecting piece of antechamber and probe, and connected to the particle size analyser. An enrichment of the stack effluent with test material was prevented by release of the main sample flow to the vicinity through a cyclonne separator toped by a filter.

11. Experimental Findings

The passage rates were investigated for the two legs, to 0 m and +90 m, including the connected aerosol monitors and grab filters.

11.1 Large Particles

The following passage rates were found for the +90 m system:
- ABB-GAD = 88.3%
- ribbon filter = 52.4%

The following passage rate were found for the 0 m system:
- ABB-GAD = 80.3%
- ribbon step filter = 67.8%

Ribbon filter as well as ribbon step filter are not very apt for this kind of measurement, as the removal of the ribbon filter after a test cannot be untertaken without losses.

11.2 Fine Particles

The following passage rates were found for the +90 m system:
- ABB-GAD = 72.5%
- ribbon filter = 57.8%

The following passage rates were found for the 0 m system:
- ABB-GAD = 52.0%
- ribbon step filter = 46.2%

The worse passage rates of ribbon and ribbon step filter can be explained by unfavourable flow paths within the devices.

12. Particles Size Distribution

Another investigation was executed to find out wether the passage rates show a significant correlation to size for the two different spectra. To this end all large particle samples were analysis just after dosing. To determine the size distribution after the pipework, the ABB-GAD was replaced by a tube with the same extraction probe connected to the scattered light particle size counting analyser (SPZ) that was used above.

12.1 Results

The particle size distribution measured at +90 m and 0 m are almost identical. They are, if compared to the original size distribution, slightly shifted to smaller sizes. This can be explained not only by the unsufficient statistics at the lower end of the size range (sample weight in the ABB-GAD app 2.5 g), but also to a lower extend by cominution of the beads, of which a considerable quantity are not ballshaped but amorphous structures.

The sizes distributions of the salt particles are nearly the same after both legs. But compared to the original size distribution an obvious shift to smaller sizes can be seen. The reason is, that the particles are subject to concussions on the pipe wall, and are this cominuted. If the shift of the size distribution is attributed to cominution of the salt cristalls, it is valid to say, that no selective particle transport occurs. This statement is proved by background aerosol measurements. The cumulative number distributions measured in the stack effluent and those encountered at the end of both legs agree favourably.

To show were the losses in the system predominantly occur, each leg has to be considered as two seperators connected in series. One separator - the grid of probes, elbows, tubes to the ABB-GAD, the ABB-GAD itself - are identical for both parial systems. The other separator, the delivery pipe, is different for both legs. If c is the concentration of the final samples and c_0 the original concentration than

$$\frac{c}{c_0} = a \cdot e^{-1m}$$

were a denotes the separation rate of the first separator common to both particle systems, 1 the length of the straight falling tube, and m the separations rate per unit length. If the calculation is done with the aforementioned results, a is found to be 0.81. In other words, the concentration is already diminished in the grid by 20%. In contrast to this, the change of concentration at +90 m is almost 90% of that behind the grid (0.81 * 0.895), which is the known result 72.5%. The concentration at 0 m is 64% of that behind the grid, i e 53% of the original concentration.

12.2 Recurrent Test

One requirement of the authorities was to provide an easy means for recurrent tests of the entire systems's proper functioning (integral calibration). To meet this requirement, the following method was proposed after several pre-tests, an validated experimentally: A solution of 2% potassiumbromide is atomised with an ultrasonic mist generator. The mist is fed directly into the effluent duct on the roof of the processing building. Samples are taken with the ABB-GADs at 0 m and +90 m. The measurement is counter-checked by also operating the emergency measurement during tests.

12.3 Results

This test procedure provides the following passage rates for large and small pariculate matter: - partial system 0 m = 38.1%
+90 m = 43.5%
- emergency measurement +120 m = 49.8%
Compared to the results of the passage rate tests with fine particles (+90 m = 72.5%; 0 m = 52.0%), it is obvious that the figures of the integral tests are considerably lower. This is caused by additional losses. During the dosage it is possible that large drops from within the feeding pipe in the duct wall, and are lost. Other reasons for losses are the reverse vortex at the duct entrance in the stack, the constructions at +90 m, and the large stack wall area.

A statistical evaluation of the samples shows that the lower figures for the ABB-GADs were not caused by unrepresentative sampling. The dispersion of the test results is 3.75% for 0 m and 3.3% for +90 m. This it holds good, by neglecting individual errors, at a statistical confidence level of 95%:
partial system 0 m = 38.1 ± 5.3%/+90 m = 43.5 ± 4.6%
This is not valid for the emergency measurements. A dispersion of 18.3% is calculated for them yielding a confidence interval of ± 25.6%. Please note that the emergency system is operated at underkinetic conditions, why it should show higher values anyway. Particle filters exposed directly to the flow are, therefore, not an adequate means for this measuring purpose.

A possible dependance of the overall passage rate upon the effluent flow rate is currently investigated.

13. Final Remarks

The system was intentionally laid out with high transport velocities, actually in large Reynolds (high turbulence) and large Stokes numbers. The high velocities chosen are not only able to transport large particles, but due to the high turbulence fine particles are moved to the duct wall so rapidly that they can hardly ad here and are rebound to the flow. Transport of particules at large Stokes and Reynolds numbers is rarely dealt with in literature, but is fully approved by the calibration tests. The new stack effluent monitoring system meets all previously demanded requirements; redundancy and testability are provided for.

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ACTIVITIES OF THE COMMISSION OF EUROPEAN COMMUNITIES IN THE FIELD OF ENVIRONMENTAL RADIOACTIVITY MONITORING

F. LUYKX, A. JANSSENS and M. DE CORT
Commission of the European Communities
DG XI.A.1 and J.R.C.-Ispra

Summary

The Euratom Treaty confers a number of duties and prerogatives to the Commission, among others in the field of environmental radioactivity. One of these prerogatives was taken up again recently, i.e. the right of access to the environmental monitoring installations in order to verify their operation (Art. 35).

In terms of Art. 36 however, the Member States have always kept the Commission informed on the levels of radioactivity which they have observed. These date are stored in the REM data base and summary reports are regularly published by the Commission. Actually, the Commission is making a major effort to facilitate and to improve the integration of national data into a consistent and representative European picture of the radiological situation.

At the level of national experts working groups, important results were booked in solving the problems of data transfer, reporting limits and the establishment of a high accuracy monitoring network, which will find acceptance in the next report on environmental radioactivity in the EC Member States 1987 - 1990.

In the meantime and in parallel, intercomparison exercises between the laboratories in the EC Member States were started to demonstrate and solve eventual differences.

1. The Euratom Treaty

Chapter III of the Euratom Treaty (1957) deals with the Health and Safety aspects of the development and growth of nuclear industries and in particular with the establishment of uniform safety standards to protect the health of workers and of the general public (Articles 30-33). These standards have been laid down in the form of Council Directives, the current provisions dating from 1980 (80/836/Euratom), with a partial amendment in 1984 (84/467/Euratom) [1].

Other articles of the Euratom Treaty (Art. 35-38) deal with the levels of radioactivity in the air, water and soil. The obligations laid down therein have not been enforced in the same way as the Basic Safety Standards, but applied directly.

The main responsibility for establishing an environmental monitoring programme lies with the Member States. Article 35 however also gives the Commission a right of access to the monitoring facilities in order to verify their operation and efficiency. The Commission has decided to take up this right of access again (December 1989), in this way meeting multiple requests of the European Parliament.

The authorities shall keep the Commission informed of levels of radioactivity to which the population is exposed, by periodically communicating the data obtained with the facilities referred to in Art. 35. The initiatives of the Commission in the field of environmental

monitoring correspondingly got the label "Art. 35/36", as opposed to the "verifications in terms of Art. 35".

Article 37 requests that Member States provide "general data relating to any plan for the disposal of radioactive waste in whatever form", and the Commission delivers an opinion on whether this is liable to result in the radioactive contamination of another Member State. The modalities of this obligation have been formulated in a Recommendation of the Commission (91/4/Euratom) and in the present context, it is interesting to note that Member States are asked to communicate every two years a statement on the radioactive waste discharges from each category 1 or 2 installation (essentially the installations of the nuclear fuel cycle, including their decommissioning).

In this way the Commission receives both data on radioactivity in the environment and on radioactive discharges, but on a different basis. A summary of discharge data for all nuclear power stations and reprocessing plants in the Community is regularly published by the Commission, together with an assessment of the corresponding population exposure. The next report will cover the period 1977-1986.

2. Verifications in terms of Art. 35

Since the right of access and verification conferred to the Commission in Art. 35 had not been exercised for about thirty years, and since nuclear industry has evolved significantly since 1957 when the Treaty was signed, the wording of the Article needed to be interpreted in the present context. The main issue was whether "environmental monitoring" pertained only to the monitoring stations installed by the authorities all over the country, irrespective of their proximity to nuclear installations. Even if stations that are part of the environmental monitoring plan of the installations were included, it is known that in normal operation, the emissions are currently so low that the levels of radioactivity in the environment, at least in air, are not detectable. The real, albeit small, radiological impact of a nuclear installation on the neighbouring population can be assessed only on the basis of discharge data. There is now consensus among all Member States that the "environment" starts at the point at which there is no further control of the releases, i.e. the stack and discharge channel. Consequently the verifications include the discharge monitoring equipment, and this requires access to the nuclear installations. However it is clear that this does not extend the Commission's prerogatives to operational aspects: the verification has a strict technical character. Also, the main purpose of the verifications is not to express an opinion on the magnitude of the levels of radioactivity or of the annual discharges. These data are communicated anyhow in terms of Art. 36 or Art. 37. If the levels would not comply with the Basic Safety Standards, the Commission could use other means of enforcement; in particular, the provisions of Art. 38, bringing ultimately the matter before the court of justice.

This is of course extremely hypothetical. In fact, if today there is a broad consensus to proceed with the verifications, it is because it is felt that these have an added value. It is not envisaged to duplicate the responsibilities of national competent authorities. The Commission's verifications will prove beneficial in terms of confidence building and communication with the public, in particular through the European Parliament.

3. Activities related to reporting

3.1. The REM data bank

Immediately following the accident at the Chernobyl nuclear power plant, the Commission of the European Communities (CEC) set up a special Task Force to re-examine all aspects of nuclear safety, from reactor design to off-site radiological consequences and emergency planning.

From the start, many of the groups into which the Task Force was divided, pointed out the need for an easily accessible way of storing the large number of environmental radioactivity measurements coming from the different countries affected by the radioactive plume.

To meet this demand, the Joint Research Centre (Ispra) set up the REM database. which brings together and stores in a uniform way, environmental radioactivity data produced in the aftermath of the Chernobyl accident. Its aim is:
- to promote the integration of this information on a European basis and so to establish a historical record of the accident and its consequences. In this way the data can be made widely available in a coherent form for scientific study and analysis.
- to streamline the various formats adopted in the EC for reporting routine environmental measurements and to prepare the CEC report on environmental radioactivity in the European Communities.

The current number of data records stored in REM is about 500.000 which are all available to external users via the network connection X.25. The information held by the bank covers data from the twelve EC Member States, as well as other European countries for both environmental samples and foodstuffs from 1984 onwards.

3.2. Report on the environmental radioactivity in the European Community

The report on the environmental radioactivity in the European Community 1984-1986 [2] informs on air concentration, deposition, river water, drinking water and milk in the form of regional and quarterly/yearly averages. For producing the tables and calculating the average values, use was made of the REM database facility. During the preparation of the report a number of problems were encountered and were discussed at the level of experts working groups:

- Environmental radioactivity data from the Member States came in many different formats: monthly and annual reports, network monitoring data, source data on diskette. Since it causes difficulties in interpretation and presentation of the information, rationalisation is needed. Also to avoid future delay in inputting the data and to minimise typing errors, the Member States were asked to send in their data in a digitised format.

  In order to further encourage this and to achieve a standardised data exchange format, an input processor, called EasyProteo, was developed at JRC-Ispra. By making use of choice tables and checking routines, entry errors are avoided or detected early to a large extent. The software also has internal utilities that can convert the inputted data into other data formats, so they still can be used by the Member State for its own purposes.

- Differences in the "less than" values reported by the Member States were caused by variations in the applied detection limits and in constraint values based on radiological considerations, which made averaging and intercomparison very difficult. To simplify the report and to make it more transparent, uniform reporting limits will be applied. This

implies that "less than" values higher than the appropriate reporting limits will be excluded from processing.

The reporting limits are based on an individual dose limit for the public (presently 1 mSv per year [3]) reduced by a factor of 1000. The reporting limit RL is then derived by:

$$RL = \frac{DL}{RF \cdot EDC \cdot CF}$$

where : DL = dose limit (= 1 mSv/year)
RF = reduction factor of the dose limit (= 1000)
EDC = effective dose coefficient in Sv/Bq
CF = yearly consumption factor per person

Table 1 contains the various values and corresponding references that were used for calculating the reporting limits.

Table 1: Calculation of the reporting limits to be used in the report on environmental radioactivity in the EC 1987-1990

Sample type	Nuclide category	EDC *[4]* (Sv/Bq)	CF	RL
Air	gross beta (based on ^{90}Sr)	$6.0\ 10^{-8}$	8030 m^3 *[5]*	$2\ 10^{-3}$ Bq/m^3
Surface water	residual beta (based on ^{90}Sr)	$3.5\ 10^{-8}$	60 l	$5\ 10^{-1}$ Bq/l
Drinking water	^{3}H	$1.6\ 10^{-11}$	600 l *[6]*	$1\ 10^{+2}$ Bq/l
	^{90}Sr	$3.5\ 10^{-8}$	"	$5\ 10^{-2}$ Bq/l
	^{137}Cs	$1.3\ 10^{-8}$	"	$1\ 10^{-1}$ Bq/l
Milk	^{3}H	$4.0\ 10^{-11}$	200 l *[6]*	$1\ 10^{+2}$ Bq/l
	^{90}Sr	$3.5\ 10^{-8}$	"	$1\ 10^{-1}$ Bq/l
	^{137}Cs	$1.3\ 10^{-8}$	"	$5\ 10^{-1}$ Bq/l
Mixed diet	^{90}Sr	$3.5\ 10^{-8}$	365 d	$8\ 10^{-2}$ Bq/d
	^{137}Cs	$1.3\ 10^{-8}$	"	$2\ 10^{-1}$ Bq/d

Since consumption of milk and dairy products differs a lot per Member State, a value of 200 l was accepted as being representative.

The geographical gradient of consumption factors is even more striking for mixed diet. To allow intercomparison of the results a "country independent measuring unit" (Bq/d) will be used to express contamination by ingesting food.

For the moment only a limited number of countries measure "complete meals". As an intermediate solution simulated "mixed diet" contamination values, derived from measurements on ingredients, taking into consideration the diet composition, will be reported.

- One of the most drastic impacts on using reporting limits is that, in non accidental conditions, many tables will show not much more than "less than reporting limit" indications. Therefore measurements from a sparse network, consisting of a small number of monitoring stations that produce highly accurate and significant values, will be

recorded to indicate any trends and to support the fact that the true values are below reporting limits.

3.3. European participation in the WHO intercomparison exercises

Besides the above mentioned difficulties in reporting of national data in a European context and the need for organising a European network for monitoring radioactivity levels actually prevailing in the environment, a study of the different sampling and measuring techniques [7] revealed an enormous diversity of analytical techniques.
The Commission took the initiative to encourage the participation European laboratories to the intercomparison exercises performed in collaboration with the International Reference Centre of the WHO at Le Vesinet.

In autumn 1991 water samples were prepared by the IRC-WHO and were sent to 14 laboratories in the twelve Member States for analysis of ^{3}H, ^{90}Sr and ^{137}Cs.

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THE FUNDAMENTALS OF THE NEW *"DIRECTIVE ON EMISSION AND IMMISSION SURVEILLANCE IN NUCLEAR PLANTS"*

Günther Meurin,
Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, Bonn, FRG

Horst Wolf,
Ministerium für Wirtschaft, Mittelstand und Technologie des Landes Nordrhein Westfalen, Düsseldorf, FRG

Summary

The 1979 "Directive on Emission and Immission Surveillance in Nuclear Plants" is being revised. The changes agreed on 12.06.1991 by a team from the republic government and the federal states (Länder) concern in particular measures for emission surveillance in nuclear plants during accidents (including the controlled depressurization of the reactor containment) and during the plant decommissioning phase, as well as measures for emission and immission surveillance in fuel cycle plants which are not nuclear power plants.

1. Legal basis for emission and immission surveillance in nuclear plants

The definitive rules on emission and immission surveillance in nuclear plants in the atomic and radiation protection regulations of the Federal Republic of Germany are §§ 46, 48 of the Radiation Protection Ordinance [1]. The surveillance principles laid down here have remained unchanged since 1976 despite other changes made to the Radiation Protection Ordinance.

1.1 Emission surveillance

If it is possible that radioactive material could escape into the air, water or ground, then care must be taken to ensure that the draining of radioactive material is monitored and specified according to type and activity (§ 46 (1) Radiation Protection Ordinance).

1.2 Immission surveillance

In the case of nuclear plants, the appropriate authorities (atomic law licensing authorities, atomic law supervisory authorities) can order the measurement of dose rates as well as the determination of the activity of samples in the environment, according to a plan of measurement to be laid down (§ 48 Radiation Protection Ordinance).

2. Current surveillance practice and reasons for updating the "Directive on Emission and Immission Surveillance in Nuclear Plants"

2.1 Current surveillance practice

To establish a common practice for the execution of §§ 46, 48 of the Radiation Protection Ordinance in the individual federal states with regard to specific features of each plant and location, the federal minister responsible for nuclear safety and radiation protection set out measures in the 1979 "Directive on Emission and Immission Surveillance in Nuclear Plants" [2] , which in total should make it possible to evaluate human exposure to radiation using limiting dose values (§§ 44, 45 Radiation Protection Ordinance) , in so far as the exposure to radiation is a result of the draining of radioactive materials with air and water from nuclear plants. (According to the federal structure laid down in the constitution (Grundgesetz) the federal states (Länder) are, with few exceptions, responsible for the execution of the atomic and radiation protection regulations, while the republic government throughout is responsible for nuclear safety and radiation protection. Since 05.06.1986 this has been the Minister for the Environment, Nature Conservation and Nuclear Safety.)

The individual measures laid down in the 1979 directive are tailor-made for the areas of emission and immission surveillance in nuclear plants which in that time were recognised as being in urgent need of regulation. From the current viewpoint this means that there are still sufficient specified measures for the

surveillance of the regulated operation of nuclear plants; on the other hand, the 1979 directive contains few concrete standards for emission/immission surveillance in nuclear fuel cycle plants which are not nuclear power plants (e.g. fuel assembly factories, intermediate fuel assembly storage, nuclear research centres). For these plants, the current practice of emission and immission surveillance is based in each case on individual decisions made by the authorities responsable under atomic regulations. This means balanced adoption to the measures agreed for nuclear plants. For future practice, the new directive should provide stricter measures, specifically oriented to each plant. This was stated as follows in the resolution passed by a team from the republic government and the federal states on 12.06.1991:

- The "Directive on Emission and Immission Surveillance in Nuclear Plants" regulates the emission and immission surveillance of plants and activities according to §§ 6, 7, 9 and 9b of the Atomic Act [3]. This directive serves the execution of §§ 46 and 48 in conjunction with §§ 36, 38, 44, and 45 of the Radiation Protection Ordinance.

- The setting of objectives and the principles of emission and immission surveillance are to be represented in a general part of the directive. Concrete measures regarding specific plants and specific activities for A (nuclear plants), B (fuel assembly factories) and C (fuel assembly intermediate and ultimate storage facilities) are contained in appendices.

2.2 Further reasons for updating the directive

Further reasons for updating the directive are based on experience with nuclear technology since 1979 (present state of the art), and the conclusions drawn from the reactor accident in Chernobyl with regard to radioactivity surveillance in the Federal Republic of Germany and the nuclear power stations in operation there. This concerns in particular the following points which were noted in the resolution passed by the government/federal states on 12.06.1991 as being in urgent need of regulation:

- Clarification of the requirements of the incident instrumentation in nuclear power stations (Nuclear Safety Standard KTA 3502 [4]) in the area of emission surveillance.

- Clarification of the requirements of an emission surveillance for events in nuclear power plants which, according to the Radiation Protection Ordinance, fall into the accident category and whose radiology effects should be limited by controlled depressurization of the reactor containment.

- Clarification of the requirements of emission and immission surveillance during the decommissioning of plants requiring licensing according to § 7 (1) of the Atomic Act (e.g. nuclear power stations), on the ground that

  -- the decommissioning and safe enclosure of such plants requires licensing (§ 7 (3) Atomic Act) and is subject to state supervision (§ 19 Atomic Act),

  -- the number of plants being decommissioned in the Federal Republic of Germany is increasing as plant designs which went into operation in the seventies will reach the end of their operating life in the near future. In addition, decommissionings are being carried out on official orders.

- Clarification of the immission surveillance of nuclear plants during incidents/accidents as distinguished from the measures set out in the Precautionary Radiological Protection Act, which was added to the atomic and radiation regulations of the Federal Republic of Germany in 1986 [5].

  This act lays down in detail the duties of the government and the federal states with regard to the large scale surveillance of radioactivity in terms of volume (e.g. air, water) and in terms of area (e.g. ground, vegetation, foodstuffs). The objective of this large scale surveillance of radioactivity is to minimize as far as possible human exposure to radiation and radioactive contamination of the environment through appropriate measures in the case of incidents with considerable radiological consequences. Such incidents can be caused by the operation of nuclear plants contrary to design. Therefore it is essential that the measures to be implemented according to the directive (orientation specific to plant and

location) and the Precautionary Radiological Protection Act (orientation specific to large scale surveillance) fit in so well with eachother that the associated protection targets are reached.

3. Example cases from the updated version of the directive

The idea for the revised version of the directive which came from the resolution passed by the republic government and the federal states on 12.06.1991, has been transferred to a set of guidelines. In its contents the structure of this set is shown in Table 1. This table may illustrate the points that have been changed essentially, have been expressed in more concrete terms or for which totally new standards for measures exist in comparison with the 1979 directive by marking out these points in bold print.

As both the time available for this conference and the scope for publication of conference contributions are limited, it is not possible to explain all the changes which have been made to the 1979 directive, therefore we will limit our explanation to two case examples from the area of emission surveillance.

3.1 Emission surveillance in nuclear power stations during incidents and accidents

For emission surveillance during incidents in nuclear power plants, the new directive will adopt the requirements of the Nuclear Safety Standard KTA 3502 "Incident instrumentation" [4]; with it, the following will become obligatory on determing measuring range end points for the "Incident Surveillance Display Equipment" and the "Wide Range Display Equipment":

- When measuring range end points are being set, the dose value of § 28 (3) of the Radiation Protection Ordinance (e.g. 50 mSv effective dose) must be taken into account. Since planning standards for the design of technical measures for protection against incidents are based among other things on these dose values, it must be possible to monitor the draining of radioactive materials corresponding to these dose values with apropriate measuring techniques.

- A defined set of incidents, according to the safety criteria and guidelines for nuclear power plants published in the Federal Legal Gazette, must determine the design of a nuclear power plant. When measuring range end points for the Incident Surveillance Display Equipment are being set, these incidents may be taken into consideration. This fulfils the principle whereupon the licensing authorities can regard the design of a plant with respect to safety during incidents as being in accordance with § 28 (3) of the Radiation Protection Ordinance, if the operating utility has used those incidents as a basis for plant design which are given here as a standard for the setting of the measuring range end points of the Incident Surveillance Display Equipment. An individual proof can be disregarded if the equipment for the surveillance of radioactive inert gases, aerosols and iodine is designed in such a way that certain measuring range end points given for the Incident Surveillance Display Equipment are maintained; the values in question here are those from Table 2 which correspond to the Nuclear Safety Standard KTA 1503.2, draft, from 23.06.1992 [6].

- The dose values given in § 28 (3) of the Radiation Protection Ordinance identify the border between incident and accident (compare appendix I to § 2 (1) of the Radiation Protection Ordinance). Thus the Wide Range Display Equipment is to be based on measuring range end points which correspond to the dose value given in § 28 (3) of the Radiation Protection Ordinance on the basis of the "Incident evaluation basis" [7]. An individual proof can be disregarded under the same requirements as mentioned above.

- By the measuring range end points of the Incident Surveillance and the Wide Range Display Equipment there is an enlarged measuring range which makes it possible to identify the emission situation from incident up to accident for a wide range. Nevertheless, additional measures for emission surveillance which must correspond to the layout requirements in the resolutions passed by the (german) Reactor Safety Commission and Radiation Protection Commission, must be implemented for the depressurization of the reactor containment. With regard to the expansion of the measuring range, this condition can be regarded as fulfilled if the equipment for the surveillance of radioactive inert gases, aerosols and iodine is so designed that the measuring range end points given for the depressurization are maintained; the values in question here also are those from Table 2 .

3.2 Emission surveillance during the decommissioning and safe enclosure of nuclear power plants

In addition to this, the directive will stipulate that the emission surveillance requirements for an operating nuclear power plant are valid after decommissioning and safe enclosure for as long as nuclear fuel, fission and activation products are still contained in the plant and the possibility of a leakage of radioactive materials into air, water or ground still exists. The surveillance of radionuclide groups or single radionuclides with continuous operating measuring procedures can, however, be disregarded provided that the drainage of radioactive material with air and water can be monitored by means of continuous sampling and discontinuous measurement, and the maintenance of maximum permissible activity emissions according to § 46 of the Radiation Protection Ordinance can be evaluated.

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[1] Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung - StrlSchV) vom 13.10.1976 (BGBl. I, S. 2905, 1977 S.184 und S. 269) in der Fassung der 2. Änderungsverordnung vom 18.05.1985 (BGBl. I, S. 943) und neu bekannt gemacht am 30.06.1989 (BGBl. I, S. 1321) und berichtigt am 16.10.1989 (BGBl. I, S. 1926), zuletzt geändert durch das Gesetz zum Einigungsvertrag vom 23.09.1990 (BGBl. II, S. 885)

[2] Bundesminister des Innern (BMI): Richtlinie zur Emissions- und Immissionsüberwachung kerntechnischer Anlagen vom 16.10.1979 (GMBl 1979, S. 668)

[3] Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren (Atomgesetz - AtG) vom 23.12.1959 (BGBl. I, S. 814) in der Fassung der Bekanntmachung vom 15.07.1985 (BGBl. I, S. 1565), zuletzt geändert durch das Gesetz zur Verbesserung der Überwachung des Außenwirtschaftsverkehrs und zum Verbot von Atomwaffen, biologischen und chemischen Waffen vom 05.11.1990 (BGBl. I, S. 428)

[4] Kerntechnischer Ausschuß (KTA), KTA-Regel 3502 (11/84): Störfallinstrumentierung, Bekanntmachung vom 06.12.1984 (Bundesanzeiger 1985 Nr. 40a)

[5] Gesetz zum vorbeugenden Schutz der Bevölkerung gegen Strahlenbelastung (Strahlenschutzvorsorgegesetz - StrVG) vom 19.12.1986 (BGBl. I, S. 2610), zuletzt geändert durch das Gesetz zum Einigungsvertrag vom 23.09.1990 (BGBl. II, S.885)

[6] Kerntechnischer Ausschuß (KTA), KTA-Regelentwurf 1503.2 (6/92): Überwachung der Ableitung radioaktiver Stoffe mit der Kaminfortluft bei Störfällen (Bekanntmachung im Bundesanzeiger in der Vorbereitung)

[7] Bundesminister des Innern (BMI): Störfallberechnungsgrundlagen für die Leitlinien des BMI zur Beurteilung der Auslegung von Kernkraftwerken mit DWR gegen Störfälle im Sinne des § 28 (3) StrlSchV vom 18. Oktober 1983 (Bundesanzeiger Nr. 245a vom 31.12.1983)

Table 2: Measuring range accumulated values in Bq/m³ which are to form the basis for the surveillance of radioactive material in air during incidents and accidents

	inert gases	**aerosols**	**gaseous iodine**
Incident Surveillance Display, vent stack	1 E 11	2 E 5	2 E 5
Wide Range Display, vent stack	1 E 13	2 E 8	2 E 8
Reactor containment depressurization	2 E 15	1 E 11	8 E 11

Table 1: Outline of the content of the revised version of the "Directive on Emission and Immission Surveillance in Nuclear Plants". (Bold print indicates essential changes to the 1979 directive and new standards)

General Part of the Directive

1. Objective
2. Emission surveillance
2.1 Principles
2.2 Operation according to regulations
2.3 Incident, accident
2.4 Decommissioning and safe enclosure of plants
2.5 Supervision of surveillance by the operating utility
3. Registration of diffusion conditions
4. Immission surveillance
4.1 Principles
4.2 Measurements prior to going into operation
4.3 Measurements during operation according to regulations
4.4 Measurements during incident, accident
4.5 Measurements during the decommissioning and safe enclosure phase
4.6 Definition of the measuring programme for loperating utilities and indepent measuring centres
4.7 Exposure paths to be surveyed
4.8 Media to be surveyed
4.8.1 Air
4.8.2 Water
4.8.3 Ground and vegetation
4.8.4 Foodstuffs
4.9 Radionuclide to be surveyed
4.10 Sampling and measurement procedures
4.11 Sampling and measurement location
4.12 Frequency of measurement and sampling
4.13 Detection limits and measuring range
5. Documentation and compilation of report
5.1 Results of emission surveillance
5.2 Results of immission surveillance

Appendix A: Nuclear Power Plants

A.1 Emission surveillance in nuclear power plants
A.1.1 Surveillance of the drainage of radioactive material with air
A.1.1.1 Operation according to regulations
A.1.1.2 Incident, accident
A.1.2 Surveillannce of the drainage of radioactive material with water
A.1.3 Surveillance of the drainage of radioactive material in the decommissioning and safe enclosure phase
A.1.4 Supervision of surveillance by the operating utility
A.2 Registration of the conditions for diffusion of radioactive material at the location of nuclear power plants
A.2.1 Diffusion of radioactive materials in air
A.2.2 Diffusion of radioactive materials in water
A.2.3 Completion of the registration period
A.3 Immission surveillance of nuclear power plants
A.3.1 Measures for off-site surveillance prior to going into operation and during operation according to the regulations
A.3.1.1 Measures implemented by the operating utility
A.3.1.2 Measures implemented by independent measuring centres
A.3.2 Measures for off-site surveillance during incidents and accidents
A.3.2.1 Measures implemented by the operating utility
A.3.2.2 Measures implemented by independent measuring centres
A.3.3 Measures for off-site surveillance in the decommissioning and safe enclosure phase
A.3.3.1 Measures implemented by the operating utility
A.3.3.2 Measures implemented by independent measuring centres
A.3.4 Procedure for the establishment of detection limits
A.3.4.1 Measurements prior to going into operation, during operation according to the regulations and in the decommissioning and safe enclosure phase
A.3.4.2 Measurements during incident, accident
A.3.5 Quality assurance by means of comparitive analysis

Appendix B: Fuel assembly factories

Plant specific conversion of the standards given in the general part of the directive as in appendix A

Appendix C: Intermediate storage of fuel elements, ultimate storage of radioactive waste

Plant specific conversion of the standards given in the general part of the directive as in appendix A

LEGAL PROTECTION AGAINST TRANSFRONTIER POLLUTION BY NUCLEAR INSTALLATIONS - SOME LEADING CASES DEMONSTRATED BY COURT DECISIONS

Dr.jur. Werner Bischof
Institut für Völkerrecht der Georg-August-Universität
Göttingen, Platz der Göttinger Sieben 5, D-3400 Göttingen

Summary

The seminar-topic "Environmental Effects of Nuclear Installations" involves different legal questions of international public law, the law of international organizations as well as of the internal national law inclusively the conflict law (private international law) in connection with radiological effects to the environment and the general public. As a part of this whole legal field of atomic energy law a short report is given on the current status of the jurisdiction of national courts on the legal remedies and recourse-rights of the citizens, especially of those living near the border, against the construction and operation of nuclear installations in a neighbouring state in respect of potential radiological emissions. A selected bibliography gives informations on the main legal literature.

1. Main Problems of Transfrontier Legal Protection

The written or unwritten (customary) public international law does not forbid the construction and operation of nuclear installations near the border of neghbouring States under the condition

- that the international generally accepted rules and standards of nuclear safety and radiation protection are observed;
- that precautions are taken for nuclear or radiation incidents and accidents;
- that the international conventions on nuclear liability (Paris Convention; Vienna Convention) are applicable.

Until now there are no special international binding rules concerning the participation of peaple living abroad in administrative procedures and of the access of private persons or corporations to court to raise an action against the administrative authorization of the construction or operation of nuclear installations (exception: Nordic environmental Agreement of 1974). The Recommendation of the OECD Council of 11th May 1976 on equal rights of access in relation to transfrontier pollution is not binding law. The EURATOM-Treaty of 1957 and the EC secondary law give no direct rights of actions to the European Court, its Cattenom decision of 22nd September 1988 based on reference of the Administrative Court of Strasbourg under Art. 150 in connection with Art. 37 EURATOM-Treaty.

The participation of foreign, neighbouring States is object of a lot of bilateral, trilateral and multilateral conventions. The unified mulitilateral system of a consulting procedure which was planned by the EC since 1975 is still not in force. That means that we find a different practice of participation and legal protection against transfrontier pollution in the European States.

2. Participation in National Licensing Procedures

As there are no specific obligations of States by international law in respect of the participation of private persons or corporations in national licensing procedures concerning nuclear installations, the States are free in regulating the procedural requirements for the legislation of practical admission of persons or bodies who have their residence abroad. The practice of some European States is illustrated by the leading cases of court decisions:

2.1. Switzerland

By decision of the Swiss Federal Court of 22nd August 1979 in re nuclear power station Leibstadt it has been stated that private persons living in the Federal Republic of Germany are legally empowered by Article 48 letter a) of the Swiss Act of Administrative Procedure to raise an action at the Swiss Federal Court or at the Federal Council. However, foreign plaintiffs could only refer to mistakes, errors or violation of Swiss law by the competent Swiss authorities and not of legal provisions and regulations of the State in which they have their residence (Europäische Grundrechte Zeitschrift 1980, p. 625).

2.2. France

In the French Republic there are different sorts of procedures in respect of nuclear installations:

- the procedure to declare the erection of a nuclear installation as "d'utilité publique";
- the procedure for the autorization of the erection and operation of nuclear installations governed by the Nuclear Installation Decree of 11th December 1963;
- the autorization of gaseous or liquid radioactive discharges by decrees of 6th November and 31st December 1974.

All theses procedures are eventually combined by enquêtes publiques.

As a mean of legal protection against such administrative acts people who are involved in their interests (- it is not necessary for an action to have a subjective right -) may raise a "recours pour excès de pouvoir" of the administrative authority (in the case of a nuclear installation "de base" the

Prime Minister has the competence) or by a special action according to the act of classified installations of 1976.

In all these administrative or judicial procedures people with transboundary residences have the right of participation and intervention. This has been confirmed by several judgements of French administrative courts especially in connection with the nuclear installations of Cattenom.

On 23rd December 1983 the Conseil d'État decided that the German town of Saarburg together with other French communities should be admitted to the procedure concerning the interventions against the decision of the prefect of the Moseldistrict "à l'enquête préalable à la déclaration d'utilité publique" for the construction of the nuclear installation of Cattenom (C.J.E.G. 1982, p. 118).

A similar standpoint has been represented by the Tribunal administratif de Strasbourg in its decision of 8th September 1986 by which the intervention of the Landkreis Neunkirchen against the execution of the arrêté interministérielle du 21 février 1986 relatif à l'autorisation de rejet d'effluents radioactifs gazeux par le centre de production nucléaire de Cattenom has been admitted (R.J.E. 1987, p. 80) (cf. also the judgement of the same Tribunal administratif de Strasbourg of 11th June 1987, Atomwirtschaft 1987, p. 395).

2.3. Federal Republic of Germany

In Germany and by the Länder Governments who are competent for the licensing procedure there was a different practice of the participation of people living abroad in the licensing procedure as in the permission to raise an action against the construction or operation of a nuclear installation to court. There were some States (Baden-Württemberg, Northrhine-Westfalia) with a more liberal standpoint and Lower Saxony with a strong conservative line; the practice of Bavaria has been governed by a more pragmative view. That was the situation until 1985/86.

By a famous decision of the Federal Administrative Court of 17th December 1986 there is now a unified and harmonized licensing practice in Germany concerning foreigners. In 1984 a resident of the Netherlands braught an action against the 1st partial license for the erection of the nuclear power plant "Emsland" at Lingen in Lower Saxony. The plaintiff lives in the Netherlands at the border in a distance of only 25 km from the site of the planned nuclear power plant. In the decision the Administrative Court of Oldenburg dismissed the action on the ground of the principle of territorial sovereignty (Territorialitätsprinzip). In its procedure decision the Administrative Court said that an action by a person living at the other side of the border in a neighbouring State is legally not possible because the administrative act of licensing a nuclear installation by the German public authority has effects only on the German territory. Therefore, the plaintiff of the Netherlands was not affected in his rights (Administrative Court of 6th February 1985, Deutsches

Verwaltungsblatt 1985, p. 802 = Nuclear Law Bulletin 40 (1987), p. 23).

The court decision has been cancelled by the judgement of the Federal Administrative Court (Bundesverwaltungsgericht) at Berlin of 17th December 1986 which remitted the case to the Oldenburg Court (Deutsches Verwaltungsblatt 1987, p. 375 = Nuclear Law Bulletin 40 (1987), p. 23). The highest German Administrative Court did not base its decision on the principle of territorial sovereignty. It said that the legal position of foreign citizens residing near the border is not substantially defined by public international law rules which leave open the question whether a foreign citizen has a right of action before administrative courts in the Federal Republic of Germany. The question of permission an action must be responded by the national German applicable administrative law, especially the Atomic Energy Act and its implementing decrees. The purpose of the Atomic Energy Act of 1959/85 (Sec. 1) does not warrant the interpretation that only domestic rights are the object of legal protection. The main purpose of the Act is the protection of life, health and property against the risks of nuclear energy and ionizing radiation. The Act assures on the other hand the implementation of international obligations in the field of the peaceful uses of nuclear energy. These objectives give reasons for extending those provisions of the Act which are express meant to protect individual rights to all individuals who might be affected irrespective of whether they are living on the German or on the other side of the border. The court said that such a right of action must be granted especially to neighbouring citizens of the European Community.

2.4. Austria

A still strong conservative practice in the permission of people living abroad to raise actions against administative acts is represented until now by the Austrian courts.

In its decision of 30th May 1969 the Austrian Administrative Court has not permitted an action which was raised by a foreign citizen against the construction of the new airport of Salzburg. The high court said that by Austrian national air law exclusively Austrian communities and neighbours have the right of complaint and to take part in the administrative procedure. People living in Bavaria in a foreign country are absolutely excluded from the procedure irrespective of their German or Austrian citizenship (Clunet 1972, p. 647).

The two decisions of the Land Court Linz and of the District Court Lembach of January 1987 concerning the reprocessing plant at Wackersdorf (FRG) are not based on the actions of foreign citizens; the plaintiffs of both procedures were Austrian inhabitants and the object of their action was an installation which should be installed in a foreign country. Both actions has been dismissed on procedural grounds of Austrian law.

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CORRELATION BETWEEN THE WORK CYCLE OF THE ALMARAZ NUCLEAR POWER PLANT AND THE TEMPORAL EVOLUTION OF RADIATION LEVELS IN ITS ECOSYSTEM

A. Baeza, M. del Río, A. Jimenez, C. Miró and J. Paniagua
Departamento de Física, Facultad de Veterinaria, Ctra. de Trujillo s/n, 10071 Cáceres (Spain).

SUMMARY

To determine the effect on the environment of airborne emissions of radionuclides from a nuclear power station, the time evolution was measured of ^{137}Cs concentrations in soil samples. These concentrations are positively correlated with airborne emissions with half-lives greater than 8 days. However, the spatial distribution of ^{137}Cs and ^{90}Sr within a circle of radius 25 km did not indicate any significantly increased concentrations in the area downwind of the station compared to other areas where the origin of the contamination is the fallout from atmospheric nuclear bomb tests.

INTRODUCTION

As a result of human activities, long-lived natural and artificial radionuclides have been released into the environment for many years. Most artificial radionuclides deposited on the ground surface come from nuclear weapon tests, although in the Chernobyl nuclear power station accident, for example, the fallout from this radioactive cloud was, in some zones, greater than that from weapon tests[1,2,3].

The present study is a part of an Environmental Radiological Surveillance Plan that we have been undertaking around on the Almaraz nuclear power station[4]. The goals are: Firstly, to study the temporal variability of ^{137}Cs radioactive concentrations in the surface layer of the soil and its possible correlation with the gaseous effluents from this nuclear plant. Secondly, to study the spatial distribution of ^{137}Cs and ^{90}Sr concentrations so as (i) to find possible increases in the radioactive concentrations in zones corresponding to the dominant winds, and (ii) to draft maps of isoactivity that will allow us to quantify the effect of future emissions in the region under study.

1. EXPERIMENTAL

The Almaraz Nuclear Power Station is situated beside the Tagus river in the province of Cáceres (Spain) 180 km west-southwest of Madrid. At a distance of 1 km, we collected bimonthly surface soil samples, begining in 1986, with

the aim of studying the temporal variability of their radioactive concentrations. We have also made an analysis of the spatial variability of these concentrations, sampling the surface soil within a radius of 25 km around the power station. Each soil sample was obtained from five subsamples, of 506 cm^2 area by 3 cm thickness, collected in an area of approximately 100x100 m^2.

Soil samples were oven-dried, and placed in 170 cm^3 plastic Petri vials or in 1000 cm^3 Marinelli beakers. For the gamma-spectrometry analysis, we used two intrinsic Ge coaxial detectors, with relative efficiencies of 13.5 and 25.6%, and FWHM of 1.79 and 1.85 keV, respectively, all for the 1.33 MeV energy. Their efficiencies were calibrated by using different ^{152}Eu standards with suitable substrates and geometries. Each detector is coupled to a 4096-channel multichannel analyzer. The ^{137}Cs concentrations were inferred from the full absorption peak count rate at 661.6 keV. Analysis of the corresponding gamma spectra was performed using the code ESPEC[5].

After the gamma spectrometry analysis, the soil samples were calcined at 600°C for 24 hours to eliminate organic matter, a preliminary step for the radiochemical separation of the strontium[6,7].

The beta activity was measured in a continuous flow gas proportional counter calibrated with a ^{90}Sr and ^{90}Y standard, with which the counting efficiency at zero thickness was 35 %. The concentration of ^{89}Sr and ^{90}Sr was determined by making several measurements on successive days after the process of chemical separation, and the counts were adjusted using the ^{90}Y growth curve and the ^{89}Sr and ^{90}Sr decay curves[8].

2. RESULTS

In Figure 1a) we show the temporal evolution of the activity of aerosols with decay period greater than 8 days emitted by the Almaraz nuclear plant,[9] together with the levels of ^{137}Cs detected in the surface layer of the soil. In the time interval represented, the most important radioactive concentrations released into the atmosphere were during 1986, being above 10^8 Bq/m^3-bimonthly. It is also during this period and part of 1987 that the greatest levels of ^{137}Cs were recorded in the soil. The temporal maxima of the radioactive concentrations of the aerosols coincide with the dates of refuelling in one of the two units in the Almaraz plant, or with radiologically significant incidents[9].

In working with highly variable empirical data, especially time series data, it is frequently necessary to smooth the data to be able to minimize the effect of irregular or random variations. By means of a simple five-period average[10] of the data of Figure 1a), we obtain Figure 1b) in which the evolution of the

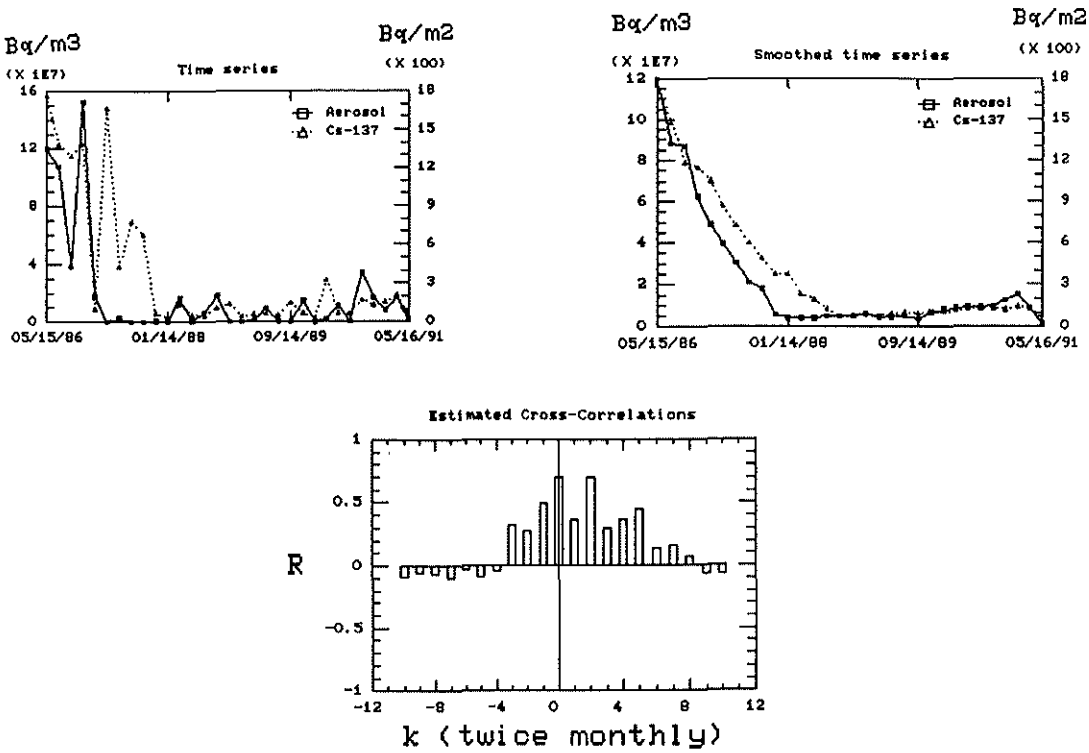

Figure 1: a) Aerosols with half-lives greater than 8 days released from the Almaraz Nuclear Power Station, and ^{137}Cs activities in soil at the Arrocampo Dam. b) Result of smoothing the above time series. c) The cross-correlation coefficient,R, between Aerosol and ^{137}Cs time series.

smoothed time series indicates that the trend of the ^{137}Cs radioactive concentrations in the soil is similar to that of the gaseous effluents. Qualitatively, this fact shows the existence of a relationship between the two series, which we can quantify by means of the cross-correlation function[10]. This estimates the correlation between one time series at time t and a second time series at time $t+k$ as a function of the lag or time differential k. It is particularly useful for determining whether two time series are correlated and, if correlated, whether one leads the other. The cross-correlation function is shown in Figure 1c). There are two maxima, one at k=0 (R=0.6967), and another at k=+2 (R=0.6964). The response to the "radioactive concentration in aerosols" time series has, therefore, two manifestations: one instantaneous, and the other lagging by two periods, or four months. There exists, then a positive correlation, at least at a local levels, between the evolution of the ^{137}Cs levels detected in the surface soil, and the maxima of gaseous evacuations produced periodically by the Almaraz nuclear plant, consequent to its functioning.

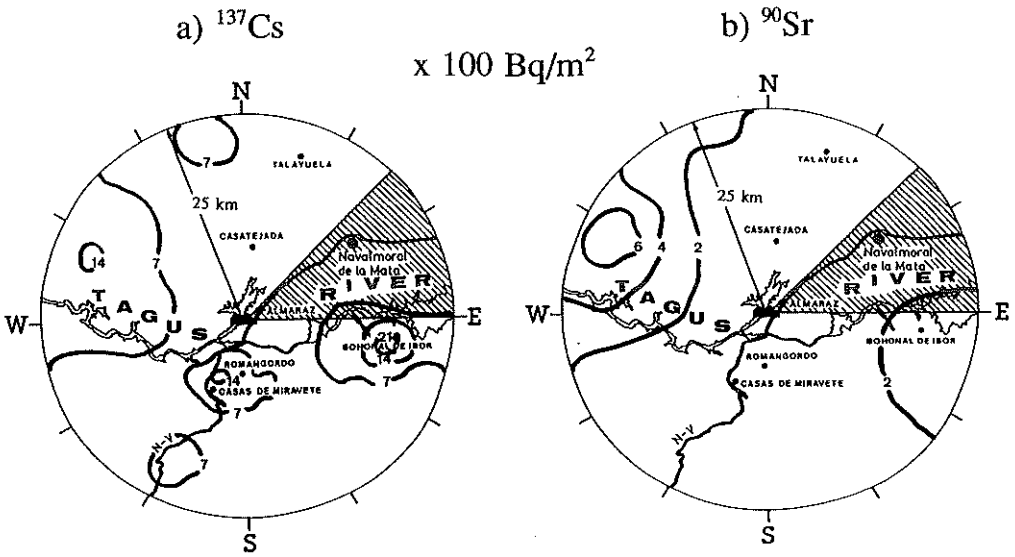

Figure 2: Spatial distribution of the radioactive concentrations, in Bq/m^2, of a) ^{137}Cs and b) ^{90}Sr around the Almaraz nuclear power plant, with indication of the zone of dominant winds, shaded area in the figures.

To calibrate the inportance of these new contaminations with respect to the radioactive concentrations from fallout from nuclear weapons tests in the atmosphere, we studied the spatial variability of the ^{137}Cs and ^{90}Sr concentrations, drawing up the corresponding isoactivity maps[11] for the zone which is potentially most sensitive to the functioning of the Almaraz nuclear plant. Figures 2a) and 2b) show those due to ^{137}Cs and to ^{90}Sr, together with the directions of dominant winds, shaded area. One notes that the cited nuclear plant is situated in soils with minimal levels of ^{137}Cs and ^{90}Sr, and also that the relative maxima of activity for both isotopes are detected in directions that do not coincide with the dominant winds in the region, which would presumibly be the direction most affected by the gaseous evacuations produced by this, nuclear plant. Thus, for distances less than 25 km from the Almaraz nuclear power plant, and given the different retention characteristics of the soils of the region, the repercussion of the gaseous emissions is negligible as against that of the fallout from atmospheric nuclear tests.

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ON-LINE MONITORING OF ALPHA AND BETA EMITTERS IN LIQUID EFFLUENTS

M. Iwatschenko-Borho, A. Gagel, R. Löw, W. Rieck
FAG Kugelfischer Georg Schäfer KGaA
Radiation Measuring Technology Division
POB 1660 W-8520 Erlangen Germany

Summary

A new kind of monitor for the quasi-continuous measurement of non-volatile alpha and beta emitters in liquid effluents has been developed. Based on a standard automatic sample changer the addition of a computer controlled evaporation unit allows the periodical preparation of samples which can be measured without the attenuation effect of the water. The fully automatic procedure results in detection limits which are several orders of magnitude lower than those of customary systems using large area proportional counters mounted above an overflow tank.

1. Introduction

Beside a few unsuccessful attempts in the early 60s (e.g. figure 1) the only commercially available monitors for the continuous surveillance of pure alpha and beta emitters in water are based on the registration of the few particles reaching a detector (GM-tube or large area proportional counter) which is in close contact to the liquid of interest. Due to the short range of charged particles in water this method is rather insensitive. Especially in the case of alpha particles only a thin layer of a few micrometers can contribute to the effective sample volume. For beta emitters calibration factors which vary by a factor of 1000 for different energies have to be considered (figure 2). In order to overcome these problems we decided to include a periodical evaporation procedure into a fully automatical monitoring system.

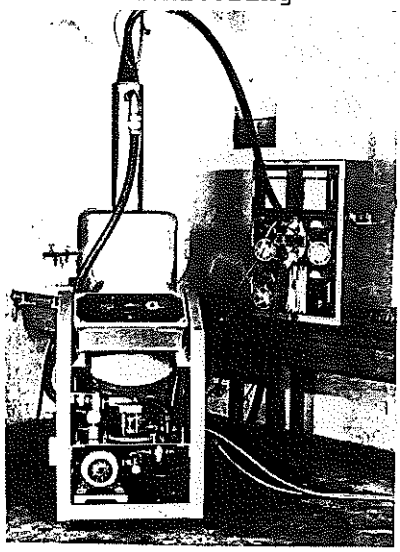

Figure 1: FHT 95 Water Monitor (≈1960). Here the water was vaporized and the α/ß-activity was measured with the help of an aerosol monitor.

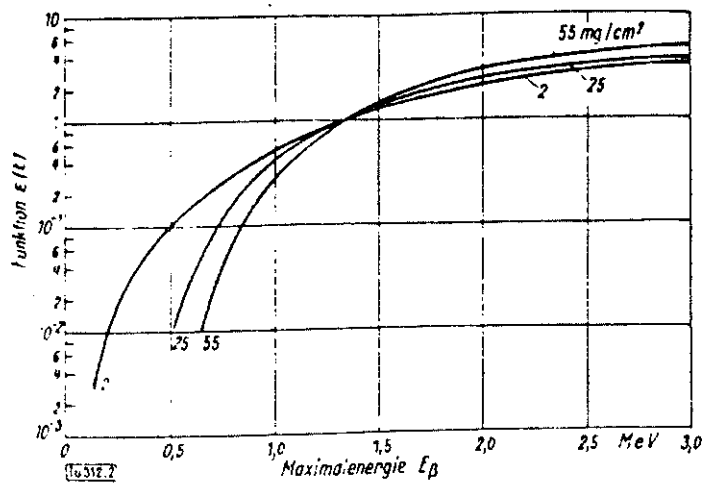

Figure 2: Direct beta monitoring in water: efficiency normalized to K-40 as a function of the maximum beta energy for different thicknesses of the detector window [1]

2. Set-Up of FHT 1900 Alpha-Beta Water Monitor

The monitoring system (figure 3) essentially consists of

a) an FHT 770 R automatic sample changer for up to 60 planchets including an FHT 7000 electronic analyzer and/or a PC

b) a sandwich type proportional counter for a gross low-level alpha/beta-measurement

c) optionally a semiconductor detector for on-line alpha spectroscopy including a vacuum chamber and a vacuum pump.

d) a water supply and quantisation system

e) an evaporation unit containing an IR-source and an IR-sensor for the detection of the reflected IR-radiation. The latter serves as an indicator for the completion of the evaporation process.

Item a) to c) of the above mentioned components are part of a common laboratoray system for the low-level measurement of e.g. wipe tests or environmental samples. The other parts were specially designed for the FHT 1900 water monitor.

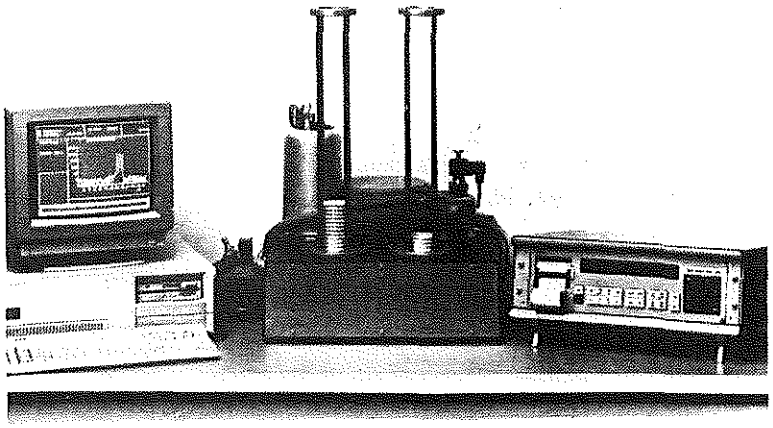

Figure 3: FHT 1900 Alpha-Beta Water Monitor (photo)

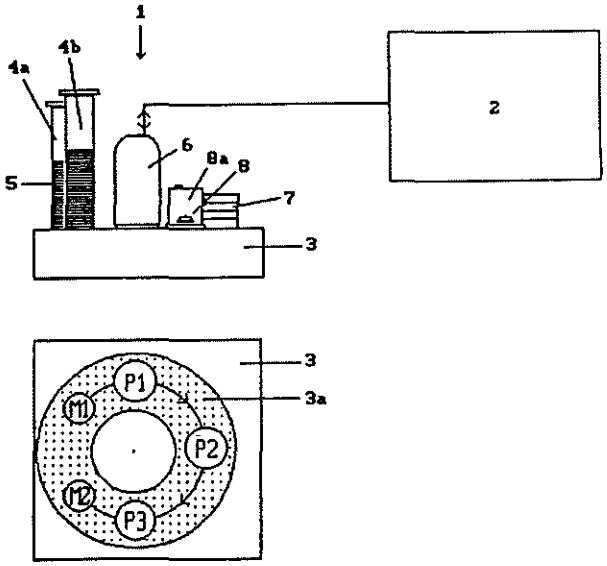

Figure 3: FHT 1900 Alpha-Beta Water Monitor (sketch)

1 FHT 1900 Alpha-Beta Water Monitor
(without electronic periphery)
2 electronical periphery (FHT 7000 or PC)
3 FHT 770 R Automatic Sample Changer
4 magazines for 60 planchets
a unused planchets (position M1)
b used planchets (position M2)
5 planchets (height 8mm, diameter 60 mm)
6 evaporation unit (position P1)
7 gross α/ß-detector with anticoincidence counter and 50 mm lead-shielding (position P2)
8 spectroscopical α-detector (position P3)
a vacuum chamber

3. Measuring sequence

During the quasi-continuous measurement of the activity concentration two different steps of the monitor have to be considered:

Step 1:
- transport of new planchet #n from magazine M1 to the evaporation unit at position P1
- transport from P1 to P2 of planchet #n-1 with the residue of the previously evaporated sample (gross α/ß-measurement)
- transport of planchet #n-2 from P2 to P3 (α-spectroscopy)
- transport of planchet #n-3 from P3 to magazine M2

Step 2:
- filling of planchet #n with water (appr. 15 ml) taken from the effluent of interest,subsequent evaporation procedure
- low-level measurements at positions P2 and P3

Step 2 (duration approximately 15 min) can be repeated several times according to the desired time resolution and the required detection limit. In cases where a short response time has to be combined with a large sampling period for each planchet short intermediate measurements can be performed between subsequent evaporation cycles.

4. Results

Using for example a total sampling time of 2 hours for each planchet a quasi-continuous recording of very low activity concentrations with detection limits of 0.1 Bq/l (α) and 0.5 Bq/l (ß) can be achieved. In this mode the stock of 60 planchets has to be filled up every 5 days. An important feature of the water monitor FHT 1900 is given by the fact that the time related variation of the activity concentration is not only recorded by the on-line data but additionally the residues in the individual planchets can be used for complementary laboratory tests in respect to a certain point of time.

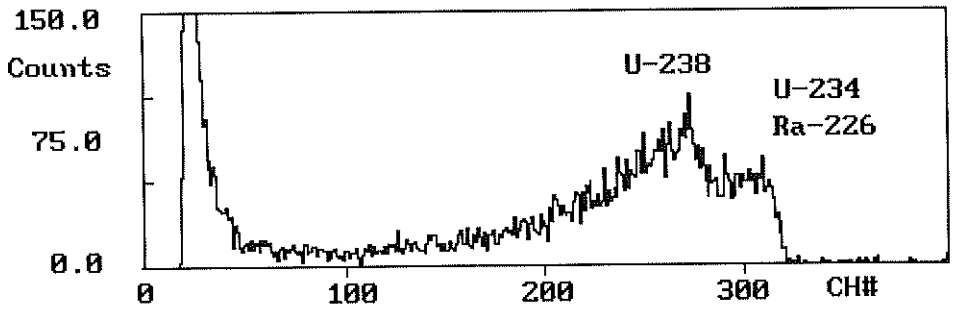

Figure 4: Alpha-spectrum of water containing a very high concentration (appr. 500 mg/l) of solid residues

The maximum quantity of water to be evaporated in one planchet is limited by the content of non-volatile elements which might lead to an increased intrinsic absorption of alpha and low energetic beta particles in the residue. Figure 4 shows an alpha-spectrum of water containing a very high concentration (appr. 500 mg/l) of solid residues. Here the marked tailing of the 4.8 MeV (Ra-226, U-234) and 4.2 MeV (U-238) peak after a total evaporation volume of 45 ml reflects the necessity of a reduced sampling time in these cases. Since the IR-reflection coefficient for the planchets covered with solid residues in excess of approximately 1 mg/cm^2 is significantly reduced, the water monitor FHT 1900 automatically reduces the sampling time and initiates the filling of a new planchet. Thus an underestimation of the alpha activity is avoided.

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RADIOACTIVE LIQUID EFFLUENT MONITORING AT THE *KARLSRUHE NUCLEAR RESEARCH CENTER* AND ESTIMATION OF THE RESULTING RADIATION EXPOSURE

K.-G. Langguth, D. Papadopoulos, M. Winter
Kernforschungszentrum Karlsruhe GmbH,Hauptabteilung Sicherheit

Abstract

The authorized values of activity releases with the liquid effluents to be observed by the Karlsruhe Nuclear Research Center have been revised by the responsible authority. The maximum permissible annual activity releases into the mains canal have been derived from the dose limits according to Sec. 45 of the Radiation Protection Ordinance by application of the General Administrative Provision relating to that section. The concept of the internal control of liquid effluents and of the discharge of individual waste water batches into the mains canal will be presented.

In a conservative approach, an effective dose equivalent of 31 μSv received by an adult can be estimated from measured activity releases with the liquid effluents from the Karlsruhe Nuclear Research Center. By contrast, an effective dose equivalent of as little as 2.4 μSv is calculated from the activity contents in drinking water and food measured within the framework of environmental monitoring.

1. Handling Radioactive Substances at the Karlsruhe Nuclear Research Center

Quite a number of institutes and facilities are located on the premises of the Karlsruhe Nuclear Research Center (KfK) where radioactive materials are being handled. The fields of work of these institutes and facilities differ greatly. They range from basic research through application oriented research devoted to nuclear engineering, the production of radioisotopes for medical applications, and they include services rendered in decontamination and conditioning of radioactive wastes with a view to repository storage. This great variety of fields of work implies that the chemical waste water of KfK may contain a broad spectrum of radionuclides. In addition, waste waters arise from shut down research reactors and from a reprocessing plant, which will be dismantled in the next decade.

The great number of radionuclides handled in approximately thirty facilities and institutes, together with a large volume of waste water of roughly 200,000 m^3 per year (approx. 80,000 m^3 of them domestic sewage, 60,000 m^3 potentially contaminated chemical waste water) call for a highly expensive technique of waste water activity monitoring.

2. Liquid Effluent Monitoring at KfK

The scheme of the effluent monitoring for radionuclides at KfK is shown in Figure 1. The measurements are all performed in a central laboratory on representative samples.

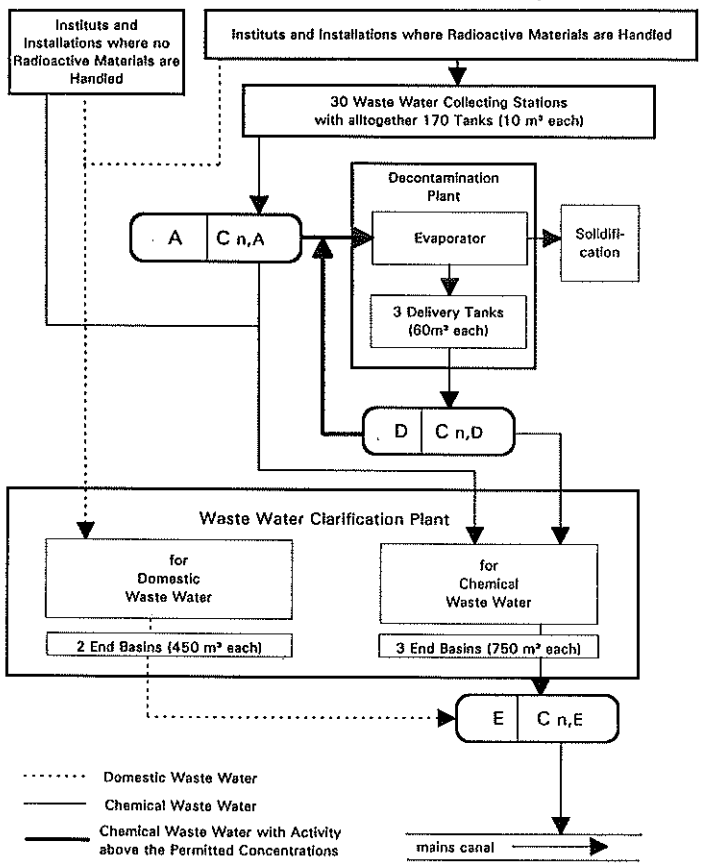

Figure 1: Flow chart of the waste water at KfK.
A,D,C: Control measurements;
$C_{n,X}$: Maximum permissible concentration for the radionuclide n.

The chemical waste water from facilities where radioactive materials are handled is first collected in tanks, which are installed in collecting stations next to the place of waste water generation. Following radioactivity measurement it is decided on the further treatment of the waste water. When the measured concentration of the radionuclide n is less than the maximum permissible concentration $C_{n,A}$, the waste water is discharged into the company owned clarification plant. When it is above that value, the waste water is transferred into the decontamination plant.

After decontamination the waste water is collected in the delivery tanks of the decontamination plant and a second measurement is made. The decision about the water is made in regard of the maximum permissible concentrations $C_{n,D}$.

In the clarification plant for chemical waste water the liquid effluents from the collecting stations and from the decontamination plant are mixed with chemical waste waters free from radioactivity, clarified in a multistage process and, finally, collected in the end basins. The concentrations of the radionuclides in the clarified waste water are determined and a decision is made in regard of the maximum permissible concentrations for discharges into the mains canal, $C_{n,E}$.

In addition, the concentrations of the activity discharges from the end basins and from the tanks of the collecting stations and the decontamination plant are

limited by application of the summation formula to the quotients of measured concentrations and maximum permissible concentrations. The sum must not exceed 1.

3. Authorized Values

Besides the maximum permissible activity concentrations for the discharge of liquid effluents from collecting stations, from the decontamination plant for liquid radioactive substances, and from the end basin of the sewage clarification plant, also the maximum permissible annual activity releases into the mains canal have been fixed in the licence granted to KfK by the authority. Until the middle of 1992 the authorized values had been defined as being multiples of the limits of the annual activity intake with water and food according to the Radiation Protection Ordinance (StrlSchV) [1]. In the revised version of that licence this dependence on the limits of the annual activity intake has been omitted. Now, the authorized values are derived from the dose limits as laid down in the Radiation Protection Ordinance.

3.1 Calculation of the Maximum Permissible Annual Activity Releases with the Waste Water

The maximum permissible annual activity releases for the radionuclide n with the waste water from the Karlsruhe Nuclear Research Center were calculated from the dose limits according to the Radiation Protection Ordinance with reference made to the Allgemeine Verwaltungsvorschrift AVV (General Administrative Provision) relating to Sec. 45, Radiation Protection Ordinance [2]. In AVV the procedures and models have been defined which have to be used in calculating the doses resulting from the discharge of radionuclides. "Dose" is always understood to mean the committed dose equivalent H_{50}.

For the exposure pathways to be taken into account for the KfK site and for the two groups of the population, namely "adults" and "infants", those amounts of activity have been calculated for each radionuclide n which, when discharged with the waste water according to the models contained in AVV, cause an exposure to radiation exactly corresponding to the limits laid down in Sec. 45 of the Radiation Protection Ordinance for the committed effective dose equivalent and for respective partial body or organ doses. The lowest activity value obtained in this way for each radionuclide n was defined to constitute the maximum permissible annual value J_n for activity releases.

In Table 1 these values J_n, the relevant group of the population (adults or infants), and the most exposed organ have been entered for those nuclides which were detected in the waste water discharged by KfK in 1991.

Table 1: Maximum permissible annual activity releases J_n of radionuclides detected in the KfK waste water discharged in 1991.

Radionuclide	J_n in Bq/a	Relevant Group of Population	Most Exposed Organ
H-3	1.5E+14	Adults	Effective
Sr-90	6.9E+09	Adults	Red Marrow
Cs-137	1.3E+10	Adults	Effective
Pu-238	1.1E+09	Adults	Bone Surface
Pu-239+240	1.9E+09	Adults	Bone Surface

Together with the liquid effluents from KfK a mix of radionuclides is discharged. Therefore, in order to be able to observe the dose limits, the activity discharge must be restricted in addition by application of the summation formula to the quotient of detected activity discharges A_n, and the values J_n. Due to specifications in the licence the sum must not exceed 2/3.

$$\Sigma A_n / J_n < 2/3$$

3.2 Derivation of the Maximum Permissible Concentrations for the Discharge of the Contents of individual Tanks

The maximum permissible concentrations for the activity discharges of individual end basin contents, $C_{n,E}$, have been limited to five times the activity concentration obtained as the quotient of J_n and the waste water volume V discharged in the reference year (1991).

$$C_{n,E} = 5 \times J_n/V.$$

By introduction of the factor 5 given time sequences in operation of the waste water systems as well as means offered by measuring technologies in the control measurements are taken into account. But still, there is no reason to fear that the $\Sigma A_n/ J_n$ will exceeded 2/3, because experience has told us that the activity concentrations in most cases are well below these values. Moreover, the discharged activity is determined nuclide specifically on the basis of weekly and monthly mixed samples, prepared in proportion of the water amounts discharged. This way the activity discharges can be limited in due course, if necesssary.

The maximum permissible concentrations of waste water discharges from the tanks of collecting stations, $C_{n,A}$ and from the delivery tanks of the decontamination plant, $C_{n,D}$, into the sewage clarification plant were specified to be a multiple of the maximum permissible concentrations of individual end basin contents:

$$C_{n,A} = 6 \times C_{n,E} \qquad C_{n,D} = 20 \times C_{n,E}$$

The factors are calculated from the mixing ratios, averaged over the year, of chemical waste waters free from activity and contaminated, respectively.

4. Radiation Exposure in the Vicinity of KfK Due to Radioactive Materials Discharged with the Liquid Effluents in 1991

The discharges of radioactive materials with the liquid effluents from KfK may cause radiation exposure of persons which live near Rheinniederungskanal, the KfK mains canal, and consume drinking water and food from that area. In calculation of the exposure one can rely either on the measured activity discharges or, more realistically, on the measured activity contents in drinking water and in food. The results of both calculations presented here make evident that the determined doses clearly underrate the dose limits as laid down in Sec. 45 of the Radiation Protection Ordinance.

4.1 Calculation of the Radiation Exposure from the Activity Discharges

The calculation was made using the RHEIN 1 computer code according to AVV. Both the effective doses and the doses for the most exposed organs have been calculated for adults and for infants living in the neighborhood of the KfK site

(see Table 2). In the calculation all exposure pathways relevant to the KfK site have been taken into account. The mean rate of waterflow in Rheinniederungskanal is 0.7 m^3 per second.

Table 2: Maximum committed dose equivalent calculated from the activities released with the KfK liquid effluents in 1991.

Activity Releases 1991		Committed Dose in μSv			
		Adults		Infants	
Nuclides	Activity in Bq	Effective Dose Equivalent	Dose of the Most Exposed Organ	Effective Dose Equivalent	Dose of the Most Exposed Organ
H-3	1.6E+13	31	-	31	-
Sr-90	3.0E+06	0.03	0.13 (RM)	0.02	0.07 (RM)
Cs-137	3.2E+06	0.08	-	0.01	-
Pu-238	7.7E+05	0.07	1.3 (BS)	0.05	0.57 (BS)
Pu-239+240	6.9E+05	0.07	1.3 (BS)	0.04	0.55 (BS)
Sum	-	31	-	31	-

(RM): red marrow; (BS): bone surface

Actual radiation exposure will certainly be well below the dose values as calculated in Table 2. The reason is that AVV has been conceived for planning purposes - implying that the models used for calculation are very conservative - in order to ensure that radiation exposure of man calculated in conformity with that provision is not underestimated.

4.2 Calculation of Radiation Exposure from the Measured Activity Contents in Drinking Water and in Food

The doses are determined as the products of ingested activities and associated dose factors. The calculation of the activity intake is based on the activity concentrations, measured within the framework of KfK environmental monitoring, in drinking water and in food from the vicinity of Rheinniederungskanal, and on the annual consumption of food and drinking water indicated in the Radiation Protection Ordinance for the reference person. The effective dose equivalents for adults calculated for 1991 is 2.4 μSv. This value is lower by a factor of 13 than the value calculated from the discharges with reference to AVV. Likewise, the clearly lower result of the dose estimated on the basis of values measured on environmental samples overestimates the actual radiation exposure because this calculation also relied on the very conservative assumptions that the population consume exclusively food and drinking water originating from the area of the mains canal.

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EMISSION LIMITS AND DOSES IN THE VICINITY OF A NUCLEAR FACILITY HAVING SEVERAL POINTS OF RELEASE

Andreas Leupin, Franz Cartier
Swiss Federal Nuclear Safety Inspectorate
CH-5232 Villigen-HSK, Switzerland

Summary

The Paul Scherrer Institute (PSI) releases radioactivity to the environment at 14 different emission points. These are spaced up to 800 m apart with varying discharge altitudes. The discharged nuclide spectrums also vary from point to point. In addition to the usual nuclides resulting from burning of radioactive waste from nuclear power plants there are some short-lived products resulting from the activation of air during accelerator work and also from isotope production. The emission limits have been established according to the relevant nuclides at the different points of release. Points with similar dispersion characteristics have been combined.

It can be demonstrated that even if the maximum permitted emissions are released, the dose remains below the recommended maximum of 0.2 mSv for persons in the vicinity of the PSI. The calculation is based on a two-dimensional dispersion model and takes into account the doses resulting from incorporation, air-immersion and the radiation from soil deposition.

1. Radioactive releases at PSI: Actual situation

At PSI gaseous releases are emitted at 14 locations which are distributed over the entire site. The following table gives a summary of these points including the important release parameters and dispersion coefficients χ_s and χ_L (cf. chapter 2):

Table 1: Emission points and dispersion coefficients

Nbr	Emission point	Stack			Height of adjacent buildings [m]	Dispersion coeff.	
		Height [m]	Diameter [m]	Velocity of discharge [m/s]		χ_s [s/m³]	χ_L [s/m³]
1	Main-stack East	70.0	1.5	26.1	-	4.6E-06	4.9E-07
2	Waste-burnig f.	25.0	0.94	6.5	13.0	2.0E-04	1.7E-05
3	Rest PSI-East						
3a	Saphir	9.5	0.8	10.0	8.35	1.5E-04	2.6E-05
3b	Proteus	15.0	0.85	2.8	appr. 15.0	8.9E-04	5.0E-04
3c	RA-waste lab	2.0	0.35	10.1	1.0	4.0E-04	9.5E-05
3d	C-lab East	7.0	2.26	2.0	5.5	7.1E-04	7.0E-05
4	Main-stack West	44.6	0.8	15.0	24.6	3.3E-05	2.6E-06
5	Double stack	24.6	0.9	4.5	24.6	2.9E-04	2.8E-05
6	Rest PSI-West						
6a	Injector II	16.2	1.22	1.7	10.9	7.1E-04	8.1E-05
6b	Tritium cabin	8.2	0.3	2.0	6.2	7.1E-04	1.2E-04
6c	ATEC	21.0	0.65	6.2	21.0	2.0E-04	2.3E-05
7a-	PET/PET-hood	13/10.5	0.28	4.0	10.0	-	-
7c	B-lab-BG	8.2	0.18	2.8	7.2	-	-

The maximum inventory available for a release at the emission points listed as number 7 is so small that we disregard these points from further investigation. Radioactive liquids are released at two discharge points to the river Aare (at PSI East and West respectively). The doses resulting from these liquid releases are small and can be neglected compared to the gaseous releases.

2. Limitation of radioactive releases and calculation of the related dose in the environment: Problems and concepts for their solution

According to the guideline "R-11" of the Swiss Federal Nuclear Safety Inspectorate, the limitation of the radioactive emissions for a nuclear installation must be chosen in such a way, that the related dose to the population in the vicinity due to that emissions remains below 0.2 mSv. The models and the proceedings we have applied to determine the emission limits in accordance to the requirement of R-11 are mentioned in the following sections. Beside the well-known problems for a plant with a single emission stack (e.g. the choice of an adequate dispersion model or the assumptions made for the wonts of the population), the main difficulty in calculating the dose in the vicinity of the PSI is the fact that one has to consider more than one emission point. Moreover, the emission points are spaced up to 800 m apart with varying stack heights and nuclide mixtures.

2.1 Dispersion coefficients and dispersion equivalent

The determination of the *dispersion coefficients* χ_L for long- and χ_S for short-term releases is mainly based on the Gauss-model described by the IAEA in [1]. This includes the consideration of plume rise Δh added to the height of the stack, the partitioning of the discharge due to the influence of adjacent buildings into a ground-level-released portion and a part released at the effective height and last but not least the effects of weather classes and the velocity of the wind. The standard deviations σ_y and σ_z which are the main parameters for the determination of the dispersion coefficients are derived from the Vogt-parameters listed in [2]. A more comprehensive compilation of the fundamental assumptions we used for our determination of the dispersion coefficients is given in [3]. The results of our determinations are presented in the last two colomns of table 1.

The models for the dispersion described in [1] and [2] are principally valid only, if the ground is "sufficiently" flat. This condition is not fullfilled for all the land in the vicinity of the PSI, but nevertheless we don't introduce any corrections for the topography. Because the hills around the PSI are largely covered with forest (i.e. they are neither inhabited nor can they be used agriculturally), this procedure is justified for calculations used only to verify the conformity of the limitations with the R-11. Nevertheless, for the determination of the actual dose to the population in the vicinity of the PSI after a non-permitted short-term release it may be necessary to take into account the influence of the topography to the deflexion of the wind field and to the dispersion of the radioactivity.

In order to enable a higher degree of flexibility to the management of the installations without affecting the radiation protection, it is usefull to unite the emission points under 3a-d and 6a-c under the name "Rest East" and "Rest West" respectively and to define a common set of limits for them. Due to the different dispersion factors this can be done by limiting the *dispersion equivalents* $Q_{eq.}^{disp.}$ which are defined as the normalization of the actual releases Q_i at the points i to a reference dispersion factor χ_R. Hence, $Q_{eq.}^{disp.} = Q_{Si} \cdot \chi_{Si}/\chi_{RS}$ for a short-term and $Q_{eq.}^{disp.} = \sum_i Q_{L,i} \cdot \chi_{Li}/\chi_{RL}$ for a long-term release. Consequently, for the points combined as "Rest East" or "Rest West", the dispersion equivalents $Q_{eq.}^{disp.}$ instead of the actual releases Q_i have to be below the limits defined in table 2.

2.2 Activity equivalent and assumed nuclide mixtures

Activity equivalents $Q_{eq.}^{act.}$ may be defined if only one dose path is of importance for a group of nuclides and if the radioactive decay during the flight-time between the stack and the affected point of interest can be neglected. These conditions are fullfilled more or less for noble gases

(important path: immersion) and for the longer lived isotopes of iodine (most important path: ingestion). The activity equivalent is calculated according to:

$$Q_{eq.}^{act.} = \sum_i Q \cdot DF_i / DF_R \ .$$

Therein DF_i is the dose(-rate)-conversion factor for the respective nuclide and DF_R the one for the reference isotope (i.e. Ar-41 for noble gases and Iodine-131 for Iodine-mixtures).
For the rest of the nuclide groups (e.g. aerosoles) it is not reasonable to define activity equivalents. Consequently, for the limitation of the release and the calculation of the related dose one has to fall back on representative "standard" nuclide mixtures. These standard mixtures have been defined for the various emission points with respect to the experience of the last years (cf. [3]).

2.3 Basic assumptions for the calculation of the dose stemming from a single emission point

The assumptions (e.g. the nutritive wonts of the people or the "standard" nuclide mixtures) and the formulas for the calculation of the effective dose at the most afflicted location in the vicinity on the basis of the emitted activity from a single stack are compiled in [4] and [5]. In addition to that , the following assumptions are made:

- All calculations are based on the ICRP-recommendations 60 as far as possible. E.g., the weighting factor for the thyroid is chosen to be 0.05.
- We take into account the entire intake of iodine from milk, vegetables and meat.
- For the immersion and for the radiation from the ground we consider a total shielding factor of 0.39 stemming from the assumption that a person is outside only 40 hours per week and that the shielding factor inside is 0.2.
- For nuclides with half-lifes smaller than 8 days we assume the transfer root-plant and plant-meat to be zero.

2.4 Superposition of the dose contribution of all emission points

To superpose the contributions of all emission points in the vicinity of the PSI, we used a grid with variable dimension and mesh-width. Both the long- and the short-term dispersion coefficients for the emission points have then been calculated for every intersection of the grid. Because the dose of a single emission point is directly proportional to the local dispersion coefficient, the contribution of every discharge point may easily be derived from the dose value at the most afflicted point (table 2). The total dose at every intersection is finally obtained by superposing all dose contributions assuming that the maximum permitted discharges are released at every emission point.

3. Results and conclusions

The limits for the releases of the single emission points have been fixed after various discussions with the responsible representatives of the PSI. Table 2 lists the limits and the related doses for an infant (1 year old) and a full-grown person for all emission points and for both a short-term release and a long-term release.
It can be demonstrated that the requirements of the R-11 are fullfilled by the fixed limits of discharge for every single emission point. Since the PSI is considered as one single plant, it has to be proved additionally that even in the worst case, i.e. if the maximum permitted emissions are released at every emission point, the superposition of the contributing doses remains below the required 0.2 mSv. The result of such dose estimations for a long-term release is presented in the figure below taking into account all paths, i.e. inhalation, ingestion, immersion and direct radiation from the soil. In many cases it may be more realistic,

Table 2: Emission limits and related doses for infants and full-grown persons

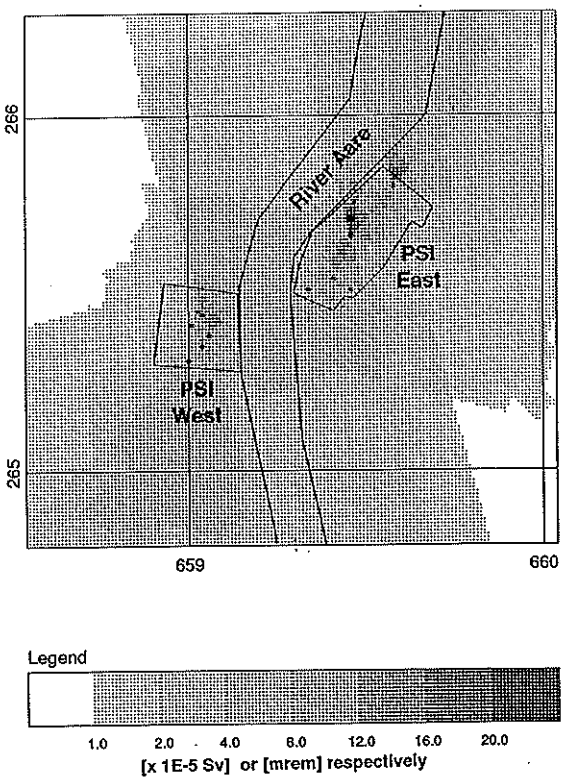
Kind of release Point of release	Short-term release Limit	Related dose [Sv] Infant	 Full-grown	Long-term release Limit	Related dose [Sv/a] Infant	 Full-grown
Gases (Ar-41-Eq.)						
1) Main-stack East	4.0E+11	1.1E-07	1.1E-07	-	-	-
2) Waste-burning fac.	4.0E+11	4.6E-06	4.6E-06	4.0E+12	1.5E-06	1.5E-06
3) Rest PSI-East	5.0E+10	1.1E-06	1.1E-06	5.0E+11	1.1E-06	1.1E-06
4) Main-stack West	2.0E+13	3.8E-05	3.8E-05	2.0E+14	1.2E-05	1.2E-05
5) Double stack	5.0E+11	8.3E-06	8.3E-06	5.0E+12	3.1E-06	3.1E-06
6) Rest PSI-West	2.0E+11	8.1E-06	8.1E-06	2.0E+12	5.4E-06	5.4E-06
Xe-122, Xe-123, Xe-125						
4) Main-stack West	2.0E+11	-	-	2.0E+12	-	-
Iodine (I-131-Eq.)						
1) Main-stack East	2.0E+09	1.3E-05	2.0E-06	3.0E+10	1.1E-05	1.6E-06
2) Waste-burning fac.	2.0E+08	5.9E-05	8.7E-06	2.0E+09	2.5E-05	3.8E-06
3) Rest PSI-East	2.0E+07	1.2E-05	1.7E-06	2.0E+08	1.4E-05	2.1E-06
4) Main-stack West	5.0E+08	2.4E-05	3.6E-06	5.0E+09	9.6E-06	1.4E-06
6) Rest PSI-West	1.0E+07	2.9E-06	4.4E-07	1.0E+08	1.7E-06	<1E-06
Aerosols ($T_{1/2}$ > 8d, β, γ, without Iodine)						
1) Main-stack East	1.0E+09	3.3E-06	2.1E-06	1.0E+10	5.9E-06	4.9E-06
2) Waste-burning fac.	1.0E+08	1.4E-05	9.1E-06	1.0E+09	2.0E-05	1.7E-05
3) Rest PSI-East	1.0E+07	2.9E-06	1.8E-06	1.0E+08	1.1E-05	9.5E-06
4) Main-stack West	2.0E+07	2.5E-06	3.9E-07	2.0E+08	3.0E-06	<1E-06
5) Double stack	5.0E+06	5.5E-06	8.6E-07	5.0E+07	8.2E-06	1.5E-06
6) Rest PSI-West	2.0E+07	1.5E-05	2.4E-06	2.0E+08	2.7E-05	5.0E-06
Aerosols (8h < $T_{1/2}$ < 8d, β, γ, without Iodine)						
4) Main-stack West	1.0E+10	2.3E-06	4.0E-07	1.0E+11	1.8E-06	<1E-06
5) Double stack	1.0E+08	2.1E-07	3.6E-08	-	-	-
α-Aerosols						
1) Main-stack East	3.0E+07	1.9E-06	2.9E-06	3.0E+08	2.4E-06	3.3E-06
2) Waste-burning fac.	5.0E+06	1.4E-05	2.1E-05	5.0E+07	1.4E-05	1.9E-05
3) Rest PSI-East	2.0E+05	1.1E-06	1.7E-06	2.0E+06	3.1E-06	4.2E-06
Tritium (tritiated water)						
1) Main-stack East	6.0E+11	1.3E-07	4.3E-08	-	-	-
2) Waste-burning fac.	4.0E+11	3.7E-06	1.3E-06	4.0E+12	3.2E-06	1.1E-06
3) Rest PSI-East	2.0E+11	1.4E-06	4.7E-07	2.0E+12	2.4E-06	<1E-06
4) Main-stack West	6.0E+12	9.2E-06	3.1E-06	6.0E+13	7.3E-06	2.5E-06
5) Double stack	1.0E+10	1.4E-07	4.6E-08	-	-	-
6) Rest PSI-West	2.0E+11	6.6E-06	2.2E-06	2.0E+12	1.1E-05	3.8E-06
Effective Dose (sum)						
1) Main-stack East		1.9E-05	7.1E-06		1.9E-05	9.9E-06
2) Waste-burning fac.		9.5E-05	4.4E-05		6.4E-05	4.2E-05
3) Rest PSI-East		1.8E-05	6.8E-06		3.2E-05	1.8E-05
4) Main-stack West		7.6E-05	4.5E-05		3.3E-05	1.6E-05
5) Double stack		1.4E-05	9.2E-06		1.2E-05	4.7E-06
6) Rest PSI-West		3.3E-05	1.3E-05		4.5E-05	1.4E-05

however, to consider only one or two of the paths (e.g. for areas which are not used for agricultural purposes).

Estimations with similar results can also be performed for short-term releases.

With such calculations it can be demonstrated that even for very unfavorable conditions the doses do the population in the vicinity of the PSI are in accordance with the requirements of the R-11, even entire permitted discharges are emitted at all emission points of the site.

FIGURE:
Dose in the vicinity of the PSI
assuming the maximum permitted release
emitted at all emission points



Superposition of the dose contributions for long-term release assuming that the maximum permitted release is emitted at all emission points (x): The most affected point (230 µSv) is situated within the enclosure of the eastern part of the PSI. Outside of the fence the doses remain below the 0.2 mSv demanded by the guideline R-11. Besides, the most affected area just around the PSI-East is neither inhabited nor agriculturally used, i.e. the estimation gives a very conservative impression of the realistic situation.
Between the two areas of the PSI one recognizes the river Aare. The distance between two coordinate lines of the map is 1 km, north is up.

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EXPERIENCE WITH THE EMISSION MONITORING AT THE *ASSE SALT MINE*

Herbert Meyer, Ingo Müller-Lyda, Rolf Stippler
GSF-Forschungszentrum für Umwelt und Gesundheit, GmbH
Institut für Tieflagerung
Theodor-Heuss-Str. 4
D-W 3300 Braunschweig

Summary

In this report the methodology of the emission monitoring at the Asse Salt Mine is described and the most important results received since start-up of the measurements are presented. The annual emissions of radioactive substances with the exhaust air are balanced with respect to their type and amount and the potential radiation exposure in the surrounding is determined. The calculated organ and effective doses are far below the limits given by the radiation protection ordinance. The measurements revealed that the concentration of HTO, C 14, and Pb 210 in the exhaust air decreases continuously, whereas that of the short-lived radon progeny remained almost constant.

1. Introduction

Since 1965 the GSF has been operating the Asse Salt Mine in order to develop and test methods for radioactive waste disposal in deep geological formations. In the course of these R&D activities a total amount of 126,000 containers with low and intermediate-level radioactive wastes has been disposed of in formerly exploited chambers in rock salt. During disposal it already became apparent that some small amounts of volatile radionuclide compounds (e.g., HTO, C 14 in carbon dioxide, and Rn 222) escape from the waste, reach the ventilation current of the mine and are exhausted into the environment. In accordance with the "Guideline on Emission and Immission Surveillance of Nuclear Plants" [1] an appropriate technique to detect and monitor the radioactive emissions with the exhaust air of the mine was installed and continuously improved.

2. Exhaust Air Monitoring Programme

The emission monitoring of the Asse Salt Mine is being performed taking into acount the specific properties of the emplaced waste following the guideline [1] and the KTA-Rule No. 1503.1. The emission monitoring can be restricted to a mere exhaust air monitoring since no sewage occurs from the mine operation. The programme has been constantly adapted to the current state of science and technology as well as recently to the new legal situation subsequent to revision of the Radiation Protection Ordinance in 1990.

3. Emission Monitoring

The ventilation of the mine is maintained by thermal draught and is supported by an additional ventilator at a rate of 30 - 100 m^3/s or $1.5 \cdot 10^9$ m^3/a, respectively (Fig. 1). The exhaust air velocity is measured continuously in the mine exhauster.

Due to the conditioning of the wastes and the standstill of emplacement operations, no major radioactive release is to be expected. Gases may escape from the waste drums into the disposal chambers, driven by barometric pressure changes in the mine atmosphere and, via diffusion, may enter into the galleries and be released with the exhaust air into the environment. These gases contain for instance Radon, volatile Tritium and Carbon compounds. The emission monitor was designed such that these radionuclides can be balanced accordingly by means of suitable collecting methods and measuring devices.

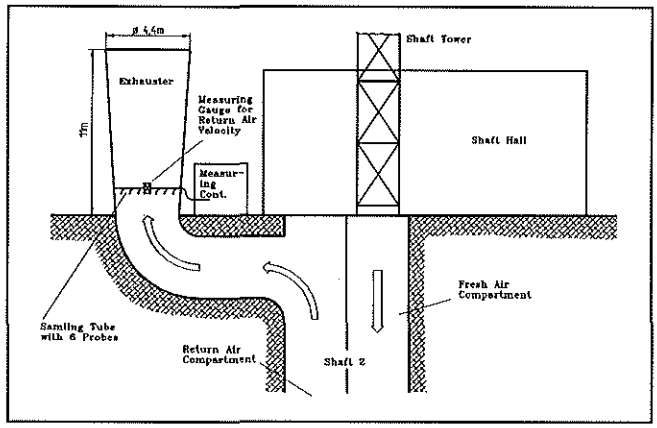


Fig. 1: Arrangement of the sampling equipment in the exhauster of the Asse Salt Mine

All relevant changes of the aerosol monitor, e.g., exchange of filters, efficiency calibrations or interruptions of the sampling rate measurement are reported to the datalogger. Moreover, the operational state of the mine ventilator, the position of the air gates in the shaft, and the exhaust air velocity as well as the meteorological data are continuously registered.

3.1 Rn 222 and its short-lived progeny

A partial air flow of about 12 m^3/h is taken from the exhaust air in the diffusor via a sampling tube (Fig. 1) according to DIN 25243, and its aerosol content is precipitated on a filter of 20 cm diameter. As comparatively high salt-dust concentrations may occur in the exhaust air the sampling rate had to be reduced so that, on the one hand, the service lives of the filters are not too short and that, on the other hand, the stipulated detection limits (40 $\mu Bq/m^3$ for Co 60) can be attained. The detection limit for the short-lived α/β-aerosol concentration is about 0.8 Bq/m^3, depending upon the salt dust content in the sampled air. With the total exhaustion rate of 100 m^3/s the detectable emission rate is about 300 kBq/h.

The α/β-contents of the deposited aerosols are measured via a commercial aerosol monitor (type LB 150 by Berthold, Wildbad) and are continuously registered by a data acquisition system (type FMA 86 by Lambrecht, Göttingen) with a sampling interval of 1 s and recording of 10-minutes mean values. They are transformed into activity concentrations using a self-developed computer software. The total α/β-concentrations determined in that way (Fig. 2) range between 30 and 300 Bq/m^3 and are conservatively declared as Rn 222. Control measurements by random sampling indicate that the Rn 222 concentrations are always smaller than the continuously measured α/β-concentrations.

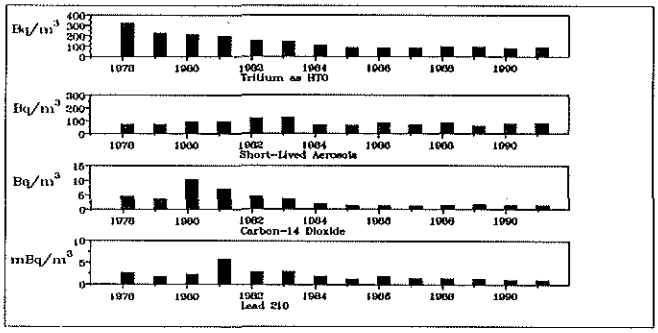


Fig. 2: Average annual concentration of radioactive substances in the exhaust air of the Asse Salt Mine

After a sampling period of 14 days the filters are changed and measured on a large-area counter. During this measurement it is made evident by the activity decline that the short-lived aerosol activity can mainly be attributed to short-lived Rn 222 progeny.

Since January 1992 continuous, integrating Rn 222 measurements using a multi-wire ionization chamber and electret dosimeters were performed throughout a number of weeks. Here, in contrast to the usual methods, the Rn 222 concentration is not measured via the decay products, but directly via the α-decay of Rn 222. The dosemeters gave an average Rn 222 concentration of 40 Bq/m^3. This average value could be confirmed by the continuous measurement, where a maximum concentration of 108 Bq/m^3 was observed. The total α/β-activity of the short-lived aerosols was twice as high during the observation period.

In the new German Radiation Protection Ordinance different limits for Rn 222 and "Rn 222 in equilibrium with its progeny" are stipulated. Thus, since 1990, for the determination of the effluents of "Equilibrium Radon" the equivalent concentration is derived directly from the continuously measured α-decay. In combination with the random sampling measurements of Rn 222 (gas) concentrations an equilibrium factor of approx. 0.6 can be obtained.

3.2 Determination of the long-lived Rn 222 progeny activity (Pb 210)

After decay of the short-lived activity, the filters for the aerosol samplers are investigated on single radionuclides using gamma-spectroscopy. Here detection limits below 20 μBq/m^3 (related to Co 60) can be achieved. The activity concentration of Pb 210 is in the range between 0.4 to 3 mBq/m^3. Other radionuclides detected in the exhaust air, such as Be 7 or Cs 137, are measured in the environmental air in somewhat higher concentrations and thus cannot be attributed to the mine emissions.

3.3 Plutonium activity

The Plutonium concentration in the exhaust air is measured semi-annually. During the sampling period of 20 days a total of 30,000 m^3 air is led through a special filter which is subsequently analyzed for Plutonium isotopes via α-spectroscopy. No Plutonium has ever been detected since the start of the measurements. Because of the high salt and soot concentrations in the exhaust air detection limits for Pu 239/240 and Pu 238 range between 0.01 to 1.6 μBq/m^3.

3.4 Tritium activity

Since the tritium concentration in the exhaust air is well below the detection level of continuously displaying measuring instruments, measurement is performed via collection processes. To determine the average HTO-activity concentration, part of the exhaust air is pumped continuously via an aerosol filter and, subsequently, through a stainless steel cylinder containing a molecular sieve. This molecular sieve fixes the moisture, carbon dioxide and hydrocarbons of the exhaust air. The air flow during one collection interval is determined by recording the number of pump strokes. The molecular sieve is removed monthly, and HTO and C 14 are measured. The activity concentration of HTO in the exhaust air (Fig. 2) ranges at approx. 100 Bq/m^3. Since tritium can also occur in the form of HT (gas), measurements were again performed in 1992 rendering a proportionate average HT-activity of 18 %. For the tritium balancing the HT-proportion is neglected. Additional to continuous collection there is also a weekly sampling to measure the moisture content in the air which is frozen out via the condensate. The average value of the weekly samples taken from the exhaust air is generally in good agreement with the monthly measured values.

3.5 Carbon-14 activity

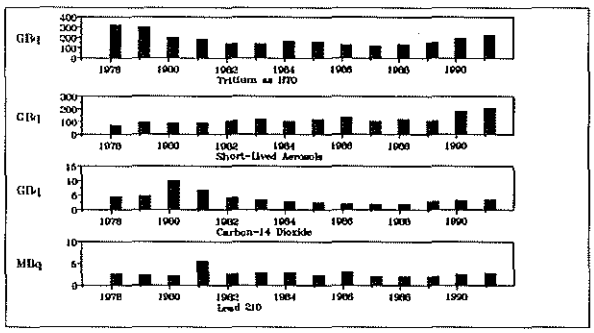
As described in Chapter 4.4 C 14 is also collected in the form of carbon dioxide during continuous sampling. C 14 is trapped by heating out the contents of the molecular sieve, led into a carrier solution and measured by means of a liquid-scintillation counter. Determination of the fraction of C 14 bound to hydrocarbon takes place at irregular intervals. This fraction amounts to approx. 10 % for all samples. Because of the radiological significance of C 14, the organic fraction is taken into consideration when calculating the total concentration and also at balancing (Fig. 2 and Fig. 3). The average C 14 total concentration in the exhaust air amounts to 1.7 Bq/m^3.

3.6 Other radioactivity

In the year 1978 the former Bundesgesundheitsamt, now BfS (Bundesamt für Strahlenschutz), investigated the exhaust air of the Asse Salt Mine on its contents of various iodine isotopes [5]. Since the random sampling measurements showed that the iodine concentration ranges well below the detection boundary of normal measuring procedures and is, therefore, to be considered as being radiologically completely negligible, a regular monitoring routine of radioiodine was omitted.

4. Results from Emission Monitoring

The measurement data obtained from radiological and meteorological monitoring are registered by a datalogger and stored in the form of 10-minute averages. They are read back by the host computer at 8-hour intervals and stored on a hard disk. The amount of short-lived aerosols is calculated by the internal software on the basis of continuous measurements of the alpha and beta impulses as well as of the total exhaust air. The daily emission rate is stored monthly on data files.



The emission values of the other nuclides are determined via the measured activity concentrations as well as from the exhaust air of corresponding sampling periods. The radionuclides emitted from the exhaust air of the mine during the past years are shown in Figures 2 and 3 [2]. The fractions of Radon, short-lived aerosols and Pb 210 already conveyed by the fresh air are not considered during balancing.

Fig. 3: Annual emission of radioactive substances of the Asse Salt Mine

5. Potential Radiation Burden on the Environment

In view of the fact that the radiation exposure in the environment due to emission is not immediately measurable, a model calculation is employed in accordance with § 45 Radiation Protection Ordinance by means of which the potential radiation burden is determined from the emission values. For this purpose the "Allgemeine Berechnungsgrundlagen" (General Calculation Fundamentals) [3] have been used until 1988. Since 1988 the Administrative Regulations [4] are applied.

The site-specific meteorological data which are required, amongst others, for the calculation of radiation exposure have been registered over a large number of years. This enables determination of the most unfavourably influenced position according to the changes of the wind direction and by means of the long-term distribution factor. The long-term distribution factor for the Asse Salt Mine was last calculated anew in 1990 by the BfS and amounted to $1.3 \cdot 10^{-4}$ s/m^3 - because of the modest height of the exhauster of only 11 m.

When determining radiation exposure only the exhaust air of the mine and the exposition pathways inhalation and ingestion are taken into consideration. Gamma and Beta submersion as well as Gamma radiation of the soil are negligible. The results indicated that the calculated doses from C 14 and Rn 222 progeny are higher for infants than for adults. Therefore small children are to be considered as the most critical members of the population. For them an effective equivalent dose of 55 μSv/a was calculated (1991). As can be seen from Fig. 4 about one fourth of the effective dose is attributable to inhalation of Rn 222 progeny. The remaining part is due to the ingestion of Pb 210. These values indicate that despite the conservative model estimates the organ doses as well as the effective doses range far below the limiting values of the Radiation Protection Ordinance.

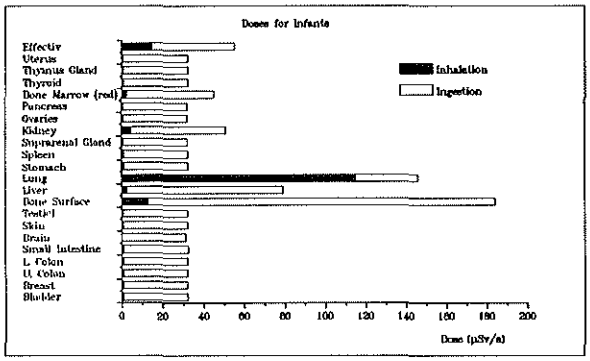


Fig. 4: Calculated organ and effective doses by emission of radioactive substances with the exhaust air of the Asse Salt Mine

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SOFTWARE-AIDED, REAL-TIME MONITORING OF ENVIRONMENTALLY RELEVANT MEASURED DATA IN NUCLEAR FACILITIES WITH LONG-TERM DOCUMENTATION AND EVALUATION

Authors:

Author: Dipl. Wirtscahftsingenieur **Hans-Peter Hamp**
Co-Author Dipl. Ingenieur **Stephan Pawlytsch**

Company:

Admintec GmbH
Bayernstraße 53
W-7530 Pforzheim
Germany

Phone: (0) 7231 / 38 08 65
Fax: (0) 7231 / 38 08 57

Summary

MessTop
Computer-aided monitoring and evaluation of measured values

MessTop is a system for long-term documentation of measured values in machine-readable form, over several years, requiring minimum space, with full measured data storage for efficient information management.

MessTop supplements or takes the place of conventional strip chart recorders. Unlike such recorders, MessTop permits the data to be represented with expanded or compressed time axis. It is also possible to superimpose otherwise independent curves in an isochronous representation.

MessTop was conceived and implemented for Brokdorf nuclear power station. It runs there on one single personal computer. The system hardware consists entirely of standardised, mass-produced components (standard PC, A-to-D converter, hard disks and MO drive etc.) with the resultant non-reliance on one single manufacturer, interchangeability of components and ensured useful life etc.

1. Benefits

The flexibility and ease of handling for an extremely wide variety of tasks in the fields of general monitoring, observation during non-stationary operating periods and in laboratory data evaluation make it possible for the user to obtain fast and clear information on all measured values. This makes it easier for him to provide precise information on sequences of events to experts and supervision authorities for instance and to document this information with graphic representations.

A measuring system such as that presented here can be used for all areas of measured data acquisition, representation and documentation, both in stand-alone mode in a control room and in large-scale installations linked to existing measuring systems.

2. Automatic integration and averaging

When curves are output on monitor and printer, the integrated measurand and the mean value are automatically generated and output for the period in question. By entering a channel-specific threshold value, it is possible to preset the datum line for integration and the mean value.

3. Tailor-made evaluations

Owing to the free configurability of the measuring channel definitions, it is possible carry out logic operations on various channels for an evaluation even subsequently. Once stored, this information is available at all times. Recurring evaluations, e.g. for annual reports, can thus be automated.

4. Fast and flexible access to measured data

Evaluations and displays/representations can be selected at any time for a freely selectable period of time without impairing measured value acquisition and limit value monitoring.
The data are accessible directly for many years since a magneto-optical disk is integrated in the archiving concept.

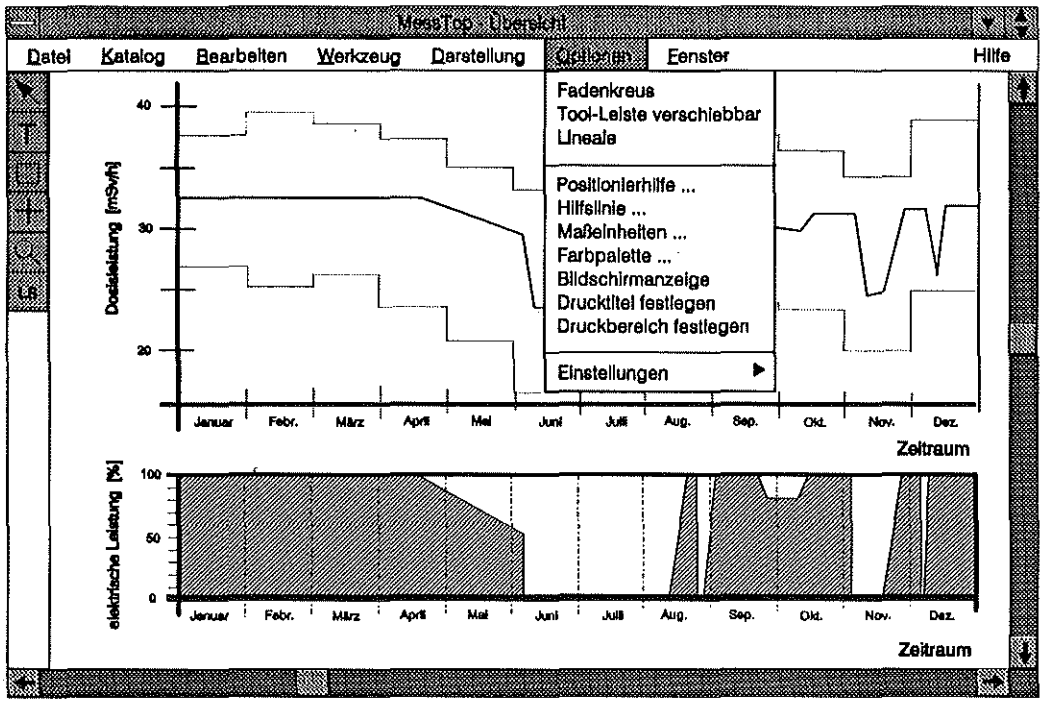

Figure1: Simultanous and isochronous on-screen comparison of individual curves, the upper curve also showing max. and min. values, apart from the average value

5. Convenient limit value monitoring for each channel.

A permitted measured value range can be defined for each channel by an upper and lower limit. The incoming measured values are monitored by the system accordingly.

A monitoring list indicates at a glance whether and when measured values of a channel have left this permitted range.
Each time an upper limit is exceeded, an actual value drops below a lower limit or an actual value re-enters the permitted range, this is recorded in a computer logbook which can be output on printer if required.

6. System features

6.1 "Raw data storage"

- Acceptance of "raw data" in the form of Volts and Amps via standardised analogue-to-digital converter boards.
- Up to 64 measuring channels can be set. Default 1/min (up to 1/s by hardware and software modifications).
- Time synchronism of the 64 channels by storage of the exact time with the "raw data".
- Storage capacity on one magneto-optical disk for a measuring period of up to 4 years.
- The "raw data" are stored as standard.
- The "raw data" are interpreted when they are output on the basis of an interpretation file which is also stored. This contains e.g. measuring range, scale (lin/log), units and limit values.

6.2 Data processing

- The scales can be changed subsequently with no conversion or accuracy losses.
- Integrated measurand formation and averaging for any periods even with logarithmic representation. Software option for noise suppression.
- Data exchange at ASCII level for further processing in other programs, e.g. in CIMPA (Canberra Software) on a PDP11/23.

6.3 Data representation

- Various curves simultaneously and isochronously or for periods to be compared in one image.
- For any periods.
- Simultaneous representation of max/min values for averaging.
- Standard forms for standard reports permit automatic report generation by entering the period under revue.
- Storage and processing software can also be integrated as a module in the Radiation-Protection Atlas.

All curve displays can be output on a laser printer.

7. Hard- and Software Requirements

- IBM Compatible Personal Computer (at least 80386)
- Microsoft Windows 3.1

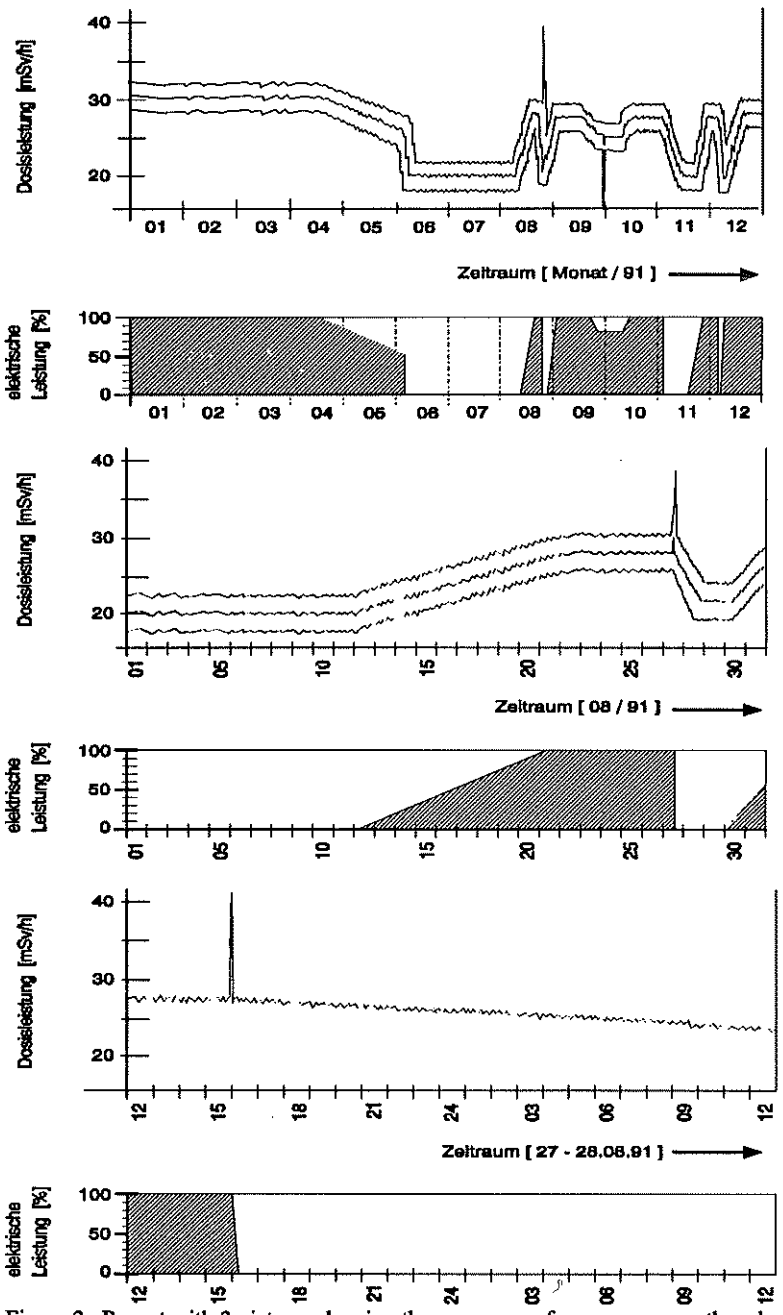

Figure 2: Report with 3 pictures showing the same curve for a year, a month and a day

IDENTIFICATION OF NON-VOLATILE IODOORGANIC COMPOUNDS IN KEROSINE FROM NUCLEAR FUEL REPROCESSING

O. Kindel[1], J. van Aarle[1], J. Furrer[3], F.-J. Herrmann[2], F.-J. Hüther[1], H.Jungclas[1],N.Psarros[1],L. Schmidt[1], P. Patzelt[1]

1). Institut für Kernchemie, Philipps-Universität Marburg
2). Wiederaufarbeitungsanlage Karlsruhe
3). Kernforschungszentrum Karlsruhe

Summary

In the kerosine from WAK, several hundred chemical compounds were detected by means of HPLC and GC. The I-129 activity was about 6000 Bq/l. After separation of the I-129 activity from the residual fission products by means of solid phase extraction, ß-measurement of HPLC fractions revealed that mainly two iodoorganic compounds were present. I-129 was enriched in the residue of vacuum distillation to 100,000 Bq/l. A further separation of the iodoorganic compounds from the other substances was achieved by reaction with trimethylamine. Quarternary ammonium salts were formed, which were identified by means of plasma desorption mass spectrometry. In the cation spectrum, the masses M=186 and M=228 were found to correspond to trimethyl n-nonyl ammonium cation and trimethyl n-dodecyl ammonium cation. Thus, iodononane and iodododecane were proved to be the non-volatile iodoorganic compounds in kerosine from nuclear fuel reprocessing.

Introduction

On burning up nuclear fuel in an atomic reactor, I-129 is produced with a cumulative fission yield of 0.757 %. The half-life of I-129 is $1.57 \cdot 10^7$ a. Therefore, it represents a potentially long-term hazard to the environment. The only major possibility for I-129 to be set free in the environment is from a nuclear fuels reprocessing plant. The I-129 concentration in the environment of the Karlsruhe nuclear fuels reprocessing plant WAK has already been investigated by several authors [1-5]. Via the food chain, this nuclide can be enriched in the thyroid glands of land-based life [6-10].

On dissolving burned up nuclear fuel during the Purex process, iodine is released mainly in elemental form. At the time of reprocessing, I-131 has already deceased due to its half-life of 8.02 d. Thus, I-129 is the only radioactive iodine isotope present. It was proved that during dissolution of burned up nuclear fuel, iodoorganic compounds are also formed [11-17]. These were reported to be mainly short-chained iodoalkanes, predominantly iodomethane. Approximately 95 % of the fission product iodine goes to the dissolver off-gas to be retained on a silver-nitrate-impregnated silica-gel filter [18]. The residual fission product iodine is partially extracted from the nitric acid phase by TBP/kerosine. It is spread through various gaseous and liquid streams in the Purex process and kerosine purification.

The main objective of the present investigation was the identification of these non-volatile iodoorganic compounds in the kerosine from nuclear fuel reprocessing.

Source of kerosine sample

The kerosine samples were taken after the second phosphate scrubbing of the kerosine purification process in the waste treatment facility. During heating of TBP/kerosine with phosphoric acid, an adduct of TBP and phosphoric acid is formed, which can be separated because of its higher specific weight. The kerosine samples, therefore, contain only 0.1-0.5 % TBP. A simplified flow scheme of WAK with kerosine purification is shown in figure 1.

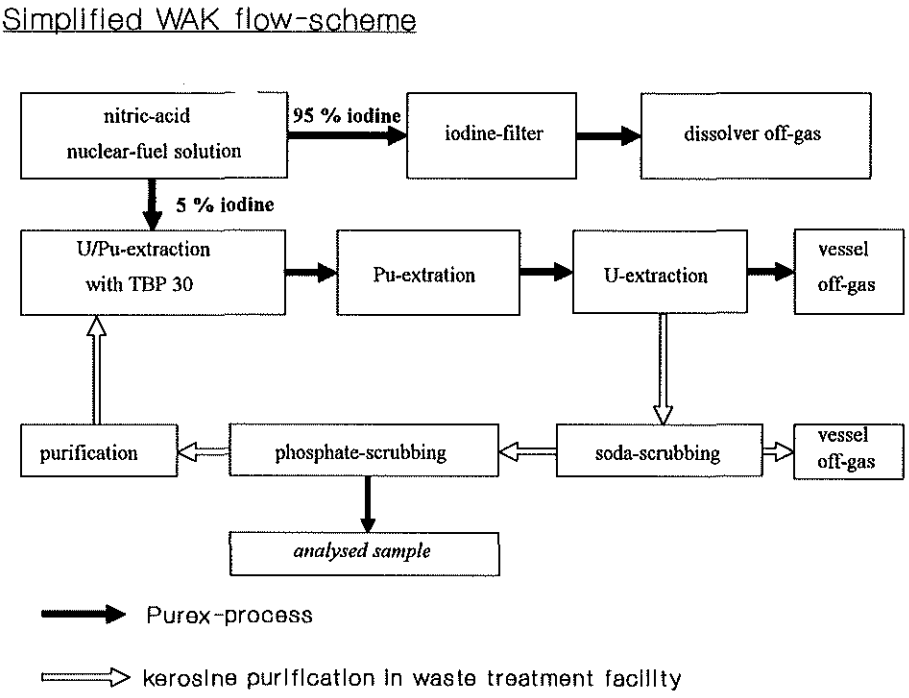

Figure 1: Simplified flow scheme of WAK with kerosine purification. The distribution of iodine to dissolver off gas and kerosine is shown. The kerosine samples were taken after phosphate scrubbing.

Properties of the samples

HPLC and GC measurements revealed that several hundred chemical compounds were present in the kerosine samples. The main component was n-dodecane. N-undecane and n-tridecane were also measured as about 10 % relative to the amount of n-dodecane. Furthermore, traces of higher n-alkanes were detected. The amount of aliphates was therefore well over 90 % of the whole sample. Separation and detection of the n-alkanes was performed by RP-HPLC (4.6*250 mm Phenospher RP20, eluent: methanol 1.0 ml/min) with refractive index detection.

The fission products in kerosine were identified by means of ß-, γ-, and x-ray spectrometry. The amount of I-129 was about 6000 Bq/l. Greater quantities of the fission products Ru-106 and Sb-125 were found than of I-129. The amounts of fission products differed from sample to sample and sometimes contained also Cs-137 and some Pu isotopes. As the ß-peaks of I-129 (E_β(max)=150 and 130 keV) were not visible in the ß-spectrum due to an at least four times higher Sb-125 activity (E_β(max)=621, 444 and 299 keV) , it was measured by means of x-ray spectrometry. Sb-125 also has x-rays in the same region, which perturbed the measurement.

Separation and identification of iodoorganic species

By means of solid phase extraction, I-129 compounds were separated from other the fission products present. A RP-NO_2 cartridge from Macherey and Nagel was conditioned with n-heptane; I-129 was selectively eluted, and the other fission products were retained on the cartridge. A kerosine sample was obtained that contained about 6000 Bq/l I-129 and only small amounts (0-10 %) of Sb-125 as contamination. In figure 2, the ß-spectra of an untreated kerosine sample and a sample purified by solid-phase extraction are shown.

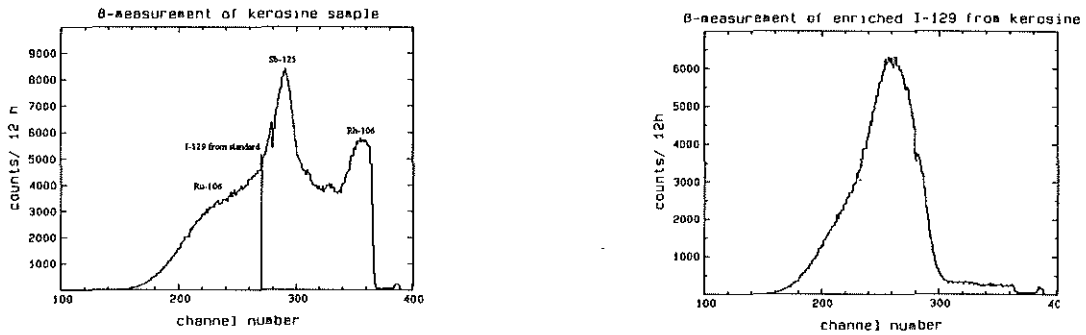

Figure 2: ß-spectra of an untreated kerosine sample (left) and a sample that was purified by solid phase extraction, which solely contains I-129 as fission product (right). Both samples contained 100 µl kerosine and 10 ml szintillator (Quicksafe A, Zinsser). The measurements were performed over a period of 12 h with an Intertechnique SL 30 ß-spectrometer.

The purified kerosine was fractionated using HPLC (4.6*250 mm Phenospher RP20, eluent: methanol 1.0 ml/min) and, after addition of 10 ml scintillator (Quicksafe A, Zinsser) to each fraction, the ß-activities were measured (Intertechnique SL 30 ß-counter). As a result, two iodoorganic compounds were found in approximately equal amounts. A histogram of the ß-counts from fractionation is shown in figure 3.

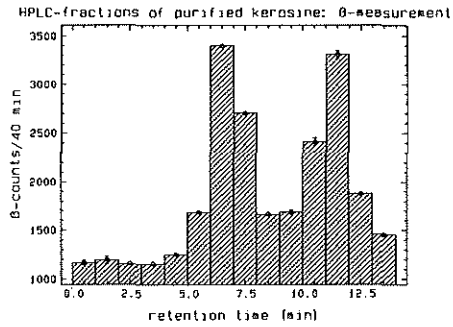

Figure 3: Kerosine sample purified by means of solid phase extraction fractionated using HPLC (4.6*250 mm Phenospher RP 20, eluent: methanol 1.0 ml/min). One fraction per min was taken. After addition of 10 ml scintillator (Quicksafe A, Zinsser), the ß-counts of each sample were measured for 40 min with an Intertechnique SL 30 ß-counter.

An enrichment of I-129 to about 100,000 Bq/l was achieved by vacuum distillation at 150°C and 10 Torr. The I-129 enriched kerosine was found in the residue. In the distillate, about 3,000 Bq I-129 were found. Fractionation of residue and distillate using HPLC (4.6*250 mm Phenospher RP 20, eluent: methanol 1.0 ml/min) with additonal ß-measurement of the fractions after adding of 10 ml scintillator (Quicksafe A, Zinsser) to each fraction revealed that the residue still contained two iodoorganic compounds, and the distillate only one. In the residue the iodoorganic species with greater retention time was enriched, and in the distillate only the iodoorganic compound with the shorter retention time was present. Thus the boiling point of the enriched compound in the residue must be considerably higher than that of n-dodecane (216.3°C). The boiling point of the other iodoorganic species which was found in the distillate must have been nearly reached during distillation. The temperature reached during distillation is near the boiling point of n-dodecane.

In the next step, iodoorganic species and kerosine were separated by means of NP-chromatography (20*250 mm Lichrosher Si60, eluent: n-heptane 4.0 ml/min). The iodine-containing fraction was dried and afterwards brought to react with a 45 % solution of trimethylamine in H_2O. This step constituted a further separation of the iodoorganic compounds from other contaminants in the kerosine. Quarternary ammonium salts were formed, by which the cations could be identified by means of plasma desorption mass spectrometry. In figure 4, the matching PDMS-cation spectrum is shown. The peaks at mass M=186 and M=228 correspond to the trimethyl n-nonyl ammonium cation and the trimethyl n-dodecyl ammonium cation.

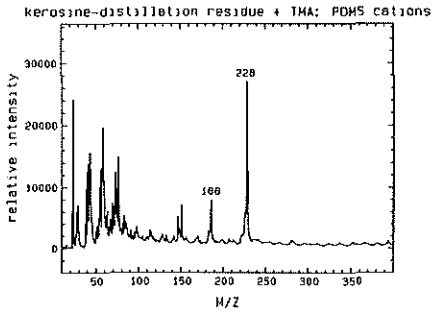

Figure 4: PDMS cation spectrum of the reaction product between kerosine distillation residue and trimethylamine, after separation from aliphates by means of HPLC and evaporation to dryness.

By means of a PDMS measurement, iodononane and iodododecane were identified as non-volatile iodoorganic species in kerosine from WAK. The measurement of these two species as standards with HPLC (4.6*250 mm Phenospher RP20, eluent: methanol 1.0 ml/min, UV-detection at 255 nm) showed the same retention times as those found by fractionation of purified kerosine (figure 3).

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Part 2: Environmental Dispersion Modelling

SYNTHESIS OF THE MODELS USED IN FRANCE FOR THE EVALUATION OF THE CONSEQUENCES OF ACCIDENTS

Bernard CRABOL

Institut de Protection et de Sûreté Nucléaire
Département de Protection de l'Environnement
et des Installations
IPSN/CEA, BP6 92265 FONTENAY-AUX-ROSES CEDEX (FRANCE)

Summary

In order to evaluate the consequences of an atmospheric release in case of an accident on a nuclear installation, different predictive models have been developed by the organisations involved in the management of the crisis. These models are of different numerical complexity: precalculated graphs, gaussian puff models or 3D models. The harmonization of these models, the definition of their use, notably in the first phases of the accident (predictive and real-time phases) have been discussed in a working group including representants of the utility, the safety authorities and the Meteorological Office. The reflexions of the group, the models already operational, those still under discussion and their use in the different technical crisis centers are presented.

I Introduction

In case of an accident on a nuclear installation, a crisis organization is settled in France in order to take the appropriate counter-measures for the protection of the population. The decision makers are adviced by teams in charge of assessing the situation of the damaged plant, performing a prognosis of the evolution of the situation and of the possible release of radionuclides in the atmosphere and finally predicting the radiological consequences of the accident. The work of these teams is helped by the use of numerical methods for simulating the dispersion of the radionuclides in the atmosphere. A great effort has been put in France since several years for the harmonization of the tools used by the different teams and on the definition of their use during the crisis. Most of this effort has been done in the framework of a group, with representants of the utility EDF, the French Meteorological Office and the Institute for Protection and Nuclear Safety -IPSN- (the technical support of the safety authorities). The work of the group consisted first to identify the actual needs of information of the authorities in the context of a crisis situation and then to choose the mathematical tools adapted to these needs. This paper gives an overview of the conclusions of this work, highlighting the reasons of the choices. Such effort of harmonization has also

been extended on an international scale and is briefly presented.

II The reflection teams of the national crisis organization.

The reflection teams for the radiological consequence assessments are distributed into three centers at the national level:

- the Emergency Crisis Center of the utility,
- the Emergency Technical Center of the IPSN,
- the Crisis Center of the French Meteorological Office,

and one at the local level, which is the crisis team of the operating utility.

During the early phase of the accident (corresponding to the off-site emergency response plan -Plan Particulier d'Intervention-), when the short term counter-measures have to be considered, the work of the utility and of the IPSN crisis teams is to predict the radiological consequences in the environment. During the later phase of the accident (corresponding to the off-site post-accident response plan - Plan Post-Accidentel-), the role of the IPSN crisis team is, according to the measurements performed in the environment, to analyse the radiological situation, notably for the longer term counter-measures.
The role of the Meteorological crisis center is to provide the other crisis centers with meteorological informations, especially, during the early phase, the forecast of the local weather, and also, if needed, at regional and long range scales.
The different crisis teams are in close contact through different channels : phone conference network, computer system terminals or fax.

III Mathematical tools used for the evaluation of the consequences in the environment

The modeling work performed in the three technical organizations involved in the crisis management, EDF, IPSN and the Meteorological Office, has led to the development of several methods which can be distinguished into three main categories:

- precalculated graphs,
- gaussian plume or puff models,
- 3 dimensional wind field models, associated with a lagrangian or eulerian diffusion model.

The need of coherence of the evaluations performed in the different crisis centers has conducted the three organizations to work together on the following objectives:

- harmonization of the methods used in the different teams,
- choice of methods adapted to the needs of the situation, according to the different phases of the crisis and the different spatial scales to be covered,
- complementarity of the work done by the teams during the crisis.

The scheme (however, not yet fully frozen), drawn hereafter and summarized on the table 1, describes the present situation concerning the calculation tools retained for operational purposes and their use during the crisis. It results for a large part from the reflection of the group, but also from the experience acquired during the numerous simulations of crisis performed in France since several years. This scheme relies on the following simple ideas which guided the work of the group:

- the purpose of the calculation tools is mainly oriented to the prediction of the consequences in the early phase of the accident for the short term counter-measures; in the later phase, as soon as contamination measurements are available, or concerning the longer term counter-measures (restriction of food commercialization or comsuption), the decisions will essentially be based on the results of the contamination measurements;
- simple tools are used when only informations with large uncertainties (at the beginning of the crisis) are available; they are run, according to the needs, in all the crisis centers, at local and national level;
- more complex tools are used when more information are known; they are run in one center, the results being communicated to the other centers.

For the consequences at local scale (approximately ten km around the site), two main methods of calculation have been developed. They are both based on the gaussian puff model using the Doury's standard deviations [1] (function of the travel time, with two classes of stability) -see remark at the end of this paper-.
During the early phase of the accident (before the beginning of the release), the main objective is to determine the zones where short-term countermeasures have to be taken. Both a prediction of the source term and the forecast of the local weather are required. In the context of such a situation, characterized by large uncertainties in the input data, a simple tool, consisting of precalculated graphs, giving values of atmospheric and surface transfer coefficients, has been adopted (see an example figure 1). These graphs correspond to different meteorological conditions (stability, wind velocity, precipitation) and are suitable for gases or aerosols. They take into account an uncertainty of ± 15° on the wind direction for simple sites (without significant topographic effect). This uncertainty corresponds to the one estimated concerning the forecast of the wind direction locally. In fact, later studies conducted by the french meteorological

office [2] have shown that it may be higher: the tests performed on some of the french nuclear sites have shown that the prediction locally of the wind direction is done with an uncertainty of ± 30° with a confidence level greater than 70 % on rather simple sites and with a lower confidence level for more complex terrain. Therefore, specific adaptations of the graphs for such sites (complex terrain) have been brought. They consist in an increase of the angle of the sector which may be reached by the plume. This increase has been evaluated, site by site, according to studies performed on the site itself or to the expert judgment of the participants of the group.

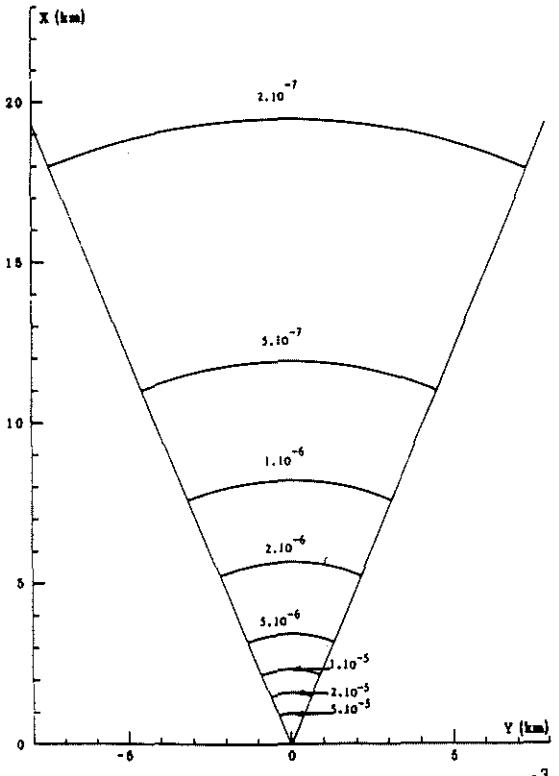

Figure 1 *: Atmospheric Transfer Coefficient ($s.m^{-3}$) Aerosols - Neutral atmosphere - Wind between 3 and 7 $m.s^{-1}$*

The second method consists in a gaussian puff model. All types of release kinetics and all types of meteorological conditions, eventually varying versus time during the release, can be taken into account.
EDF has developed, in the framework of the GEEE project, a version of such code adapted to real time assessment. This code is run locally (on the accidented site); it takes into account the meteorological data measured every ten minutes on the site and the radionuclides release rate measured at the stack. In case of necessity, the results of such real time assesssments can be sent to the national crisis centers by one of the link mentioned hereabove.
A version of such code, but not operating with on-line measurements, Sirocco [3], is also available at the IPSN crisis center.

For regional and long distances, two complementary methods have been developed: gaussian puff models applied on trajectories considered as the locations of the mass centers of pollution and 3D models. The first type of model permits a rapid estimation of the consequences and can be used during

the first phase of the accident, in order to rapidly judge the possible consequences at mesoscale and long distance, notably concerning the neighbour countries. In the later phases of the accident, when more informations are available, a 3D model can be run and gives in some cases(1) more accurate results.
A model of the first type has been developed by IPSN. The puffs follow the trajectories generated by the models of the french meteorological office every three or six hours. The concentration and doses are computed, at different times, for each point of a grid of maximum 160 x 160 points. Such model Sirocco-LD is used at the IPSN crisis center. A version with a smaller mesh size for regional problem (the code Sirocco-MD) is under development.
Concerning the second type of models, for long range transportation, the french meteorological office has developed two 3D models for the analysis or forecast of the wind, temperature,... fields [5] : Emeraude covering the whole atmosphere with a grid mesh of 1.5° in latitude and 2° in longitude and Péridot covering western Europe with an horizontal resolution of 35 km x 35 km. The wind fields generated by the models permit to calculate trajectories of transportation of pollutant. Also, an eulerian diffusion model, Média, has been developed and, coupled with the calculated wind field, permits to determine the concentration field resulting from a pollutant release. The results can then be transmitted to the other crisis centers.

For regional scale problems, different 3D prognosis models, with a smaller mesh size (of the order of 5 km) are available, while not yet ready for operational purposes: one developed at EDF (Hermes), one by the Meteorological Office (Péridot-meso), one by the IPSN (Adrea) in collaboration with the greek institute NCRS. They all permit a forecast of wind and turbulence fields over areas exhibiting complex topography, as well as the prediction of the transport-diffusion of an atmospheric release. For practical reasons, linked with the necessity of knowledge of the conditions, calculated by the Meteorological Office, at the boundary of the domain of interest, the model of this last organization will probably be retained and the results transmitted, when needed, to the other crisis centers. However, a preliminary exercice will be soon performed in order to compare the results of the wind field given by the three models (in the case where important divergences will be noticed between the results, studies would continue in order to understand the reasons for them).

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(1) the moderation in the words comes from the recent international Atmes exercice [4] which concluded that there was no obvious link between the complexity of the model and the accuracy of its prediction.

Remark: in the framework of the french-german commission, a collaborative work has been done in order to develop a common model for short distances to be used in both countries in case of emergency. A new set of standard-deviations, to be used in a gaussian puff model, has been elaborated. These standard-deviations will replace those presently used. A full presentation of this work is given in an other paper of this seminar [6].

IV Conclusion

In the framework of the french crisis organization, a great effort has been put on the harmonization of the work performed by the different teams involved in the crisis, notably on the methods used to simulate the atmospheric dispersion in case of accident. Therefore, a working group with representants of the utility EDF, the Meteorological Office and the IPSN (technical support of the safety authorities) has been set up. The work of the group consisted first to identify the actual needs of information of the authorities in the context of a crisis situation and then to choose the mathematical tools adapted to these needs.

The scheme of the situation, resulting from the work of the group, but also from the experience acquired thanks to the simulations of crisis performed in France, can now be drawn :

- a simple tool, consisting in site-specific precalculated graphs, is used when only informations with large uncertainties (at the beginning of the crisis) are available, in order to orientate the short-term counter-measures; this tool is run, according to the needs, in all the crisis centers, at local and national level;
- more complex tools (notably, gaussian puff model for real time assessment or 3D prognosis models) are used when more informations are known; they are run in one center, the results being communicated to the other emergency centers via different links, computer system terminals or fax.

Work is still to be done, notably concerning the meso-scale dispersion evaluations. Moreover, the new set of standard-deviations, developed in the framework of the french-german commission in order to harmonize the evaluations performed on both sides of the border, have to be tested and integrated in the methods for short distance consequence evaluations; such work would have to be extended, when needed, to other neighbour contries. Finally, the whole organization described hereabove, and particularly the links between the different technical emergency centers, still need tests; they will be done at the occasion of the future crisis simulation exercices to be performed in France.

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TABLE 1 : synthesis of the methods for the calculation of atmospheric dispersion used in the different technical emergency centers in France

	METHOD	OBJECT
IPSN emergency center	precalculated graphs	consequence evaluation for short term counter-measurements
	gaussian puff type models	consequence evaluation from local to long range distances
EDF national emergency center	precalculated graphs	consequence evaluation for short term counter-measurements
EDF local emergency center	precalculated graphs	consequence evaluation for short term counter-measurements
	gaussian puff type model	real time consequence evaluation at short range
Meteorological Office emergency center	3D prognosis models	weather forecast
		meso-scale and long range transport-diffusion calculation

HARMONIZATION OF FRENCH AND GERMAN CALCULATION PROCEDURES FOR ATMOSPHERIC DISPERSION FOLLOWING ACCIDENTAL RELEASES FROM NUCLEAR POWER PLANTS

K. Maßmeyer, R. Martens
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Schwertnergasse 1, D-5000 Köln 1

K. Nester
Kernforschungszentrum Karlsruhe, IMK, Postfach 3640, D-7500 Karlsruhe

H. Schnadt
TÜV Rheinland, Postfach 10 17 50, D-5000 Köln 1

B. Crabol, E. Romeo
CEA-IPSN, B.P. No. 6, F-92260 Fontenay-aux-Roses, Cedex

Summary

In case of an accident in a nuclear power plant near the French-German border different schemes for dispersion calculations in both countries will currently be applied. An intercomparison of these schemes initiated from the German-French Commission for the safety of nuclear installations (DFK) revealed in some meteorological situations large differences in the resulting concentrations for radionuclides.

An ad hoc working group was installed by the DFK with the mandate to analyse the reasons for the different model results and also to consider new theoretical concepts. The working group has agreed to apply a Gaussian puff model for emergency response calculations. The results of the model - based on turbulence parameterization via similarity approach or spectral theory - have been compared with tracer experiments for different emission heights and atmospheric stability regimes. As a reference the old modelling approaches have been included in the study. The simulations with the similarity approach and the spectral theory show a slightly better agreement to the measured concentration data than the schemes used in the past. Instead of diffusion categories both new approaches allow a continuous characterization of the atmospheric dispersion conditions. Because the spectral approach incorporates the sampling time of the meteorological data as an adjustable parameter, thereby offering the possibility to adjust the dispersion model to different emission scenarios, this turbulence parameterization scheme will be foreseen as the basis for a joint French-German puff model.

1. Introduction

The assessment of radiological consequences resulting from accidental releases of a nuclear power plant is based on the calculation of atmospheric transport of radionuclides. An intercomparison of the calculation procedures applied in Germany and France shows differences characterized as follows:

- the relevant guideline in Germany [1] recommends application of the Gaussian plume model and dispersion parameters derived from the dispersion experiments carried out at the research centers of Jülich and Karlsruhe. These parameters are appropriate for release times of about 30 minutes.
- official guidelines to calculate the radiological consequences do not exist in France. At the Institute de Protection et de Sûreté Nucleaire (IPSN) radiological consequence assessment can be done by using three different procedures which are based on Gaussian models (nomograms, plume or puff model), applying the dispersion parameters after Doury [2].

Under the mandate of the German-French Commission for the safety of nuclear installations, an intercomparison of these two methods had been carried out in 1987. Source distances up to 20 km and a release for 30 minutes at a constant rate had been considered. The results revealed differences in the magnitude of the maximum surface concentrations as well as in the location of the maximum surface concentrations. Those differences could be attributed to the different formulations of the dispersion parameters. In case of an accident at a nuclear power plant located near the French-German border, the application of both methods could result in different emergency actions in both countries. Therefore the advisory board of the DFK gave another mandate to review and update the procedures for calculation of atmospheric transport of radionuclides to come up with a calculation scheme for emergency actions which should be used in both countries.

2. Update of the modeling concepts

In october 1987 a workshop of the DFK (1988) dealt with a discussion of appropriate methods for calculating atmospheric dispersion for emergency actions focussing on source distances up to 20 km. The French and German participants agreed that a Gaussian puff model should be used to calculate the atmospheric dispersion following a nuclear accident. Different ways to determine the diffusion parameters have been discussed. A spectral approach has been favoured by the French side, while a combined similarity and convective scaling approach has been given preference by the German participants.

In contrast to the parameterization schemes used up to now the new approaches are based on boundary layer parameters like friction velocity, convective scaling velocity, boundary layer height and roughness length. These parameters can be derived from site specific meteorological measurements like temperature gradient and wind speed. They enable a calculation of dispersion parameters independent of a turbulence classification scheme like the Pasquill-Gifford scheme. A detailed description of the theories has been given in [3].

3. Comparison of the parameterization schemes

3.1 Turbulence parameters

The methods of turbulence parameterization by similarity approach and by spectral theory have been compared for typical meteorological conditions in France and Germany, averaging times of the meteorological parameters of $T = 0.5$ h and travel times t up to some 1000 s. For source heights ranging from 10 m to 180 m above ground and for unstable and neutral atmospheric stability both methods show similar results. Typical deviations between the dispersion parameters are of the order of some 10 %. Under stable conditions, however, the horizontal dispersion parameters derived from the spectral approach are much larger than the dispersion parameters calculated by the similarity method. These differences can be attributed to the large scale part of the turbulence spectrum (e.g. meandering of the windfield) which is not included in the similarity approach. Excluding the large scale part from the turbulence spectrum the spectral theory yields dispersion parameters comparable to the similarity approach [3].

3.2 Timeintegrated surface concentration

The decision whether the updated modelling techniques are superior to those used in the past could be supported by a comparison of measured and calculated tracer concentrations based on dispersion experiments, especially those which have been the basis of the old parameterization schemes. Therefore the dispersion experiments carried out at the Karlsruhe Nuclear Research Center (KfK) and additionally experiments under low wind speed conditions performed by the NOAA at Oak Ridge, Tennessee have been chosen for this purpose. For both experimental series the measured and calculated timeintegrated surface concentrations have been compared using different performance

measures [3]. For short only the intercomparison of the models with the data of the KfK-experiments will be reported here. The results of the intercomparison of the spectral approach with NOAA-data have been published in a report by Romeo [4].

Thomas and Nester (1985) [5] have summarized the results of the tracer experiments at the Karlsruhe Nuclear Research Center. The test field consisted of open spaces and built-up as well as wooded areas. The dispersion experiments comprised source heights of 60, 100, 160, and 195 m. Besides the measured concentrations and the emission data comprehensive information about the meteorological conditions up to a height of 200 m had been available. During most of the experiments two tracers have been released from different heights in several time intervals. The sampling time has been 0.5 h. For 39 sampling periods the old and new modelling approaches have been compared with the measurements.

Generally the application of different performance measures result in a similar rating of the ability of a model to calculate tracer distributions specific for each dispersion experiment. Typical examples of the performance of the new parametrization techniques can be given by an intercomparison of measured and calculated time integrated tracer-concentrations for 60 m emission height. The dispersion experiments have been grouped in 3 different stability regimes (unstable - diffusion categories A and B, neutral - diffusion categories C and D, stable - diffusion categories E and F). Only those receptor points have been retained in the datasets where the measured tracer concentrations exceed 1 % of the maximum concentration detected during each experiment. The resulting datasets comprised 112 to 167 receptor points per stability regime. They have been evaluated using the bootstrap resampling method [6]. As performance measures the fractional bias (FB) and the normalized mean square error (NMSE) have been used, the first giving an estimate of the models ability to describe high concentration values, the latter one as a global indicator for the scatter between observations and predictions. Means and confidence intervals of the two performance measures have been calculated. The results are summarized in fig. 1 and 2. The evaluation of the experiments during stable stratification has been omitted due to large deviations between modelled and measured data.

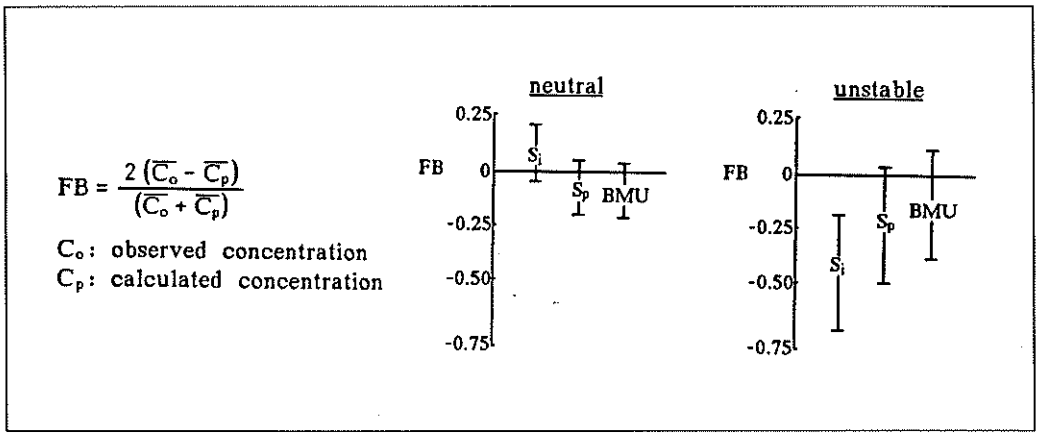

Fig. 1: Fractional bias (FB) and confidence limits (95 %-percentile) between measured and calculated concentration data at Karlsruhe Nuclear Research Center for 60 m emission height (Si = similarity approach, Sp = spectral approach, BMU = German Guideline [1])

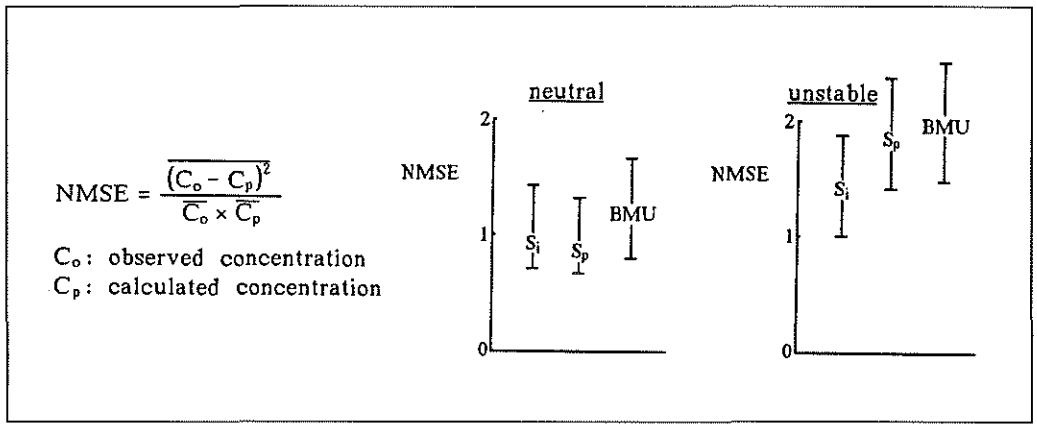

Fig. 2: Normalized mean square error (NMSE) and confidence limits (95 %-percentile) between measured and calculated concentration data at Karlsruhe Nuclear Research Center for 60 m emission height (Si = similarity approach, Sp = spectral approach, BMU = German guideline [1])

The evaluation of the experiments under unstable and neutral conditions shows that both new approaches give better results than the model of Doury (not included in fig. 1 and 2). The similarity approach and the spectral approach implemented in a puff model give results of the same quality as the calculations with the Gaussian plume model [1]. In contrast to the plume model - using dispersion parameters which have partly been deduced from the KfK-experiments - the new approaches work on the basis of turbulence parameterizations which use only site specific topographical (roughness length) and meteorological information (e. g. temperature gradient, wind speed).

However for stable stratification both new techniques fail in the prediction of the concentrations. During most of the experiments under stable conditions broad distributions of tracer concentrations have been observed at ground level. This plume geometry could not be reproduced by similarity or spectral approach. Additional calculations with the puff model using actual turbulence information in emission height (horizontal and vertical wind direction fluctuations as detected by vector wind vanes) or wind shear information using a Lagrangean particle model did not give significantly better results. Mesoscale circulation systems like meanders - attributable to the large scale turbulence - have often been reported as a mechanism for broadening of tracer plumes. They can be excluded as a reason for the resulting concentration field because time dependent wind direction data have been used for the dispersion calculations. No large scale periodicity could be detected in the time series of the wind direction in 60 m height above the ground. We assume that under stable stratification local flows have developed near the ground (due to the inhomogeneity of the experimental area and the slightly uneven terrain) which are responsible for the detected broad concentration distributions.

4. Conclusions

It can be summarized that the dispersion calculations using the similarity approach and the spectral theory show a slightly better agreement to the measured concentration data than the schemes used in the past. Both approaches only use site specific meteorological data to calculate the necessary input parameters by conventional boundary layer theory. The main advantage as compared to the older parameterization schemes is the independance from the dispersion experiments analysed. The dispersion parameters can easily be determined from temperature gradient and wind speed measurements. They allow a continuous characterization of the atmospheric dispersion independent of diffusion categories.

The spectral approach incorporates the sampling time of the meteorological data as an adjustable parameter. This allows an adjustment of the dispersion model to different emission scenarios (short release, long term release). Therefore this turbulence parameterization scheme is proposed as the basis for a joint French-German puff model. However the ability of the spectral approach to handle short term releases adequately has additionally to be tested against field data. The future work will concentrate on the implementation of the model at a nuclear site, to run it with real time data and to analyse the results in order to show its operationality.

In the context of a workshop entitled "Objectives for next generation of practical short-range atmospheric dispersion models" Olesen has pointed out that considering the situation in Europe models have been developed in an unnecessarily disorganized manner. We are so far from having an optimum rate of technology transfer [7]. Our results which have been discussed during this workshop may also be seen as a contribution to an international harmonization of calculation procedures aiming at an easier use of each others products and results.

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COMPARATIVE CALCULATIONS OF GROUND-LEVEL CONCENTRATIONS AND DEPOSITION PATTERNS RESULTING FROM ACCIDENTAL RELEAESE OF AEROSOLS USING A GAUSSIAN PLUME MODEL AND A PARTICLE SIMULATION MODEL

H. P. Berg
Bundesamt für Strahlenschutz, Postfach 10 01 49, D-3320 Salzgitter

F. Lange, R. Martens
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Postfach 10 16 50, D-5000 Köln 1

Summary

The dispersion and the deposition of large particles (up to 60 µm AED) released via the diffuser of the ventilation system of a deep-underground waste repository has been investigated and discussed using a Gaussian plume model and a particle simulation model.

1. Introduction

Safety analysis studies performed for deep-underground radioactive waste repositories indicate that accidental impacts on waste containers during implacement operations may leed to releases of radioactive particles up to 60 µm AED (aerodynamic equivalent diameter). Subsequently the particles may be released into the environment via the diffuser of the ventilation system of the deep-underground waste repository.

The Gaussian plume dispersion model as prescribed in the Incident Calculation Bases (Störfallberechnungsgrundlagen [1]) may be applied with certain limitations to describe the environmental dispersion and deposition processes of such large particles by using appropriately modified deposition velocities. The applicability of this approach, however, is limited and does not describe e.g. the continuous decline of the plume center line due to gravitational settling as the particulate cloud moves on. As a consequence, the location of the maximum ground-level concentration in the atmosphere and on the ground surface are in this modeling approach independent from the sedimentation velocity of aerosol particles - a behaviour which is not in agreement with experimental observations.

The atmospheric dispersion parameters used in the Gaussian plume model of the Incident Calculation Bases are determined by a simple numerical relationship: $\sigma_i = p_i\, x^{q_i}$, (i=y, z), i.e. the dispersion parameters σ_y and σ_z increase with source distance. The coefficients p_i and exponents q_i have been derived for different source heights and stability categories. They are based on more than 200 field experiments conducted at the research centres in Karlsruhe and Jülich. These sites are characterized by level terrain and typical roughness lengths of z_0 = 1.2 m. The coefficients p_i and q_i are tabulated in the Incident Calculation Bases. They should be applied only for sites topographically similar to Karlsruhe and Jülich. The stability classes range from very unstable (category A) to very stable atmospheric dispersion conditions (category F). - The wind speed at release height H has to be calculated according to $u(H) = u_{10}\,(H/z_{10})^m$ with z_{10} = 10 m. The Incident Calculation Bases prescribe the values for the wind speed u_{10} at 10 m height above ground and the wind profile exponent m for each stabiltity class. The values for u_{10} are chosen in such a way that they are less than a large percentage of experimentally observed wind speeds at 10 m height. Because of the 1/u-relationship of the atmospheric concentration calculated with the Gaussian plume formula this selection of u_{10} - along with the other prescribed parameters (m, p_i und q_i) given in the Incident Calculation Bases - results in a conservative estimate of ground-level concentrations in the atmosphere.

In contrast to the Gaussian plume model, particle simulation models do generally account for more complex atmospheric dispersion and deposition conditions. Deposition and sedimentation processes can be fairly easily implemented in this modeling approach by using process-specific physical models. For this study the particle simulation model LASAT has been applied in a version for flat terrain including a module for the calculation of dry and wet deposition [2]. The particle simulation algorithm allows for considering vertical profiles of horizontal wind speed u(z), standard deviations of turbulent velocities $\sigma_i(z)$, and Lagrangean time-scales $T_{Li}(z)$, (i = u,v,w). The meteorological input data required are provided by a simple atmospheric boundary layer model [3], [4]. This boundary layer model is based on state-of-the-art formulae for these vertical profiles [5], [6], [7]. The equations depend upon boundary layer parameters like Monin-Obuchov-length L, mixing layer thickness h_{mix}, and roughness length z_0. This kind of turbulence parametrization can be applied to different dispersion conditions in a continuous way without the restriction of a limited number of stability classes.

2. Calculational procedure

As an example the behaviour of large particles (up to 60 µm AED) released at a height of 45 m has been investigated. In order to examine how the sedimentation of these large particles affects the dispersion as well as dry and wet deposition when using the Gaussian plume model and the particle simulation model LASAT respectively a three step approach has been used: First comparative calculations have been carried out for particles which do not show gravitational settling and deposition. In a second step the dispersion and deposition of 60 µm-particles has been regarded. Finally for a broad size distribution of aerosol particles atmospheric dispersion and deposition have been compared for the two modeling approaches.

2.1 Dispersion of particles without gravitational settling

Comparative calculations have been carried out using a material which does not show gravitational settling and deposition. Due to the fact that the Gaussian plume model and the particle simulation model LASAT use different approaches for the parametrization of atmospheric turbulence six appropriate sets of boundary layer parameters L, h_{mix}, and z_0 have been determined being representative for the six diffusion categories prescribed in the Incident Calculation Bases. These representative sets of parameters are shown in Table 1 for a roughness length of z_0 = 1.2 m which is assumed to be typical for the site of the deep-underground waste repository. Table 1 includes the ranges of values for the Monin-Obuchov-length L and the mixing layer height h_{mix} which are related to the six diffusion categories.

Table 1: Values for Monin-Obuchov-length L and mixing layer height h_{mix} depending upon diffusion category (z_0 = 1.2 m)

Diffusion-category	Monin-Obuchov-length L		Mixing layer height h_{mix}	
	average [m]	range [m]	average [m]	range [m]
A	-15	> -22	1400	>1200
B	-40	-22 ... -61	1100	1000 ... 1200
C	-130	-61 ... -260	1000	800 ... 1100
D	infinite	-260 ... +260	800	600 ... 1000
E	130	+260 ... +61	250	100 ... 800
F	40	< +61	100	< 250

As to the down-wind distance and the height of maximum concentration both models, LASAT and the Gaussian plume model, show a reasonably good agreement for the unstable (A, B, C), neutral (D) and stable (E) diffusion categories. For the very stable diffusion category F, however, the plume center line concentration calculated by LASAT is higher than the Gaussian plume model predictions by a factor of 5. Accordingly LASAT predicts a plume width significantly smaller than for the Gaussian plume model.

2.2 Dispersion and deposition of large aerosol particles

In a second step ground-level atmospheric concentration and deposition patterns have been calculated for releases of large particles, i.e. particles with a diameter of 60 µm AED. Under these conditions the motion of particles in a turbulent atmosphere is not only influenced by the mean wind and turbulent eddies but also by gravitational settling (sedimentation velocity of approximately 10 cm/s). The calculations have been performed for the same conditions as for particles without sedimentation using the Gaussian plume model on the one hand and the particle simulation approach on the other hand.

As to wet deposition which has to be considered for the stability categories C, D, and E within the frame of the Incident Calculation Bases a good agreement has been found. Regarding dry deposition both models, LASAT and Gaussian plume model, give similar results for unstable (A-C) and neutral categories (D). For stable atmospheric conditions (E, F), however, the sedimentation process of the 60 µm-particles becomes more and more important. Under stable conditions the standard deviations of the turbulent velocity components σ_i (i = u, v, w) are of the same order of magnitude or less than the sedimentation velocity. Under these conditions on average the particles are continuously moving downwards in spite of turbulent eddies that potentially can push particles upwards. This behaviour is schematically shown in Fig.1 representing lateral views of 100 particle traces emitted from 45 m height under stable conditions. Fig.1a shows a narrow plume for simulation particles without sedimentation. No particle reaches the ground surface within source distances less than 600 m. In contrast to this Fig.1b represents the same situation for 60 µm-particles. All 100 particles are deposited on the ground within a source distance of 600 m.

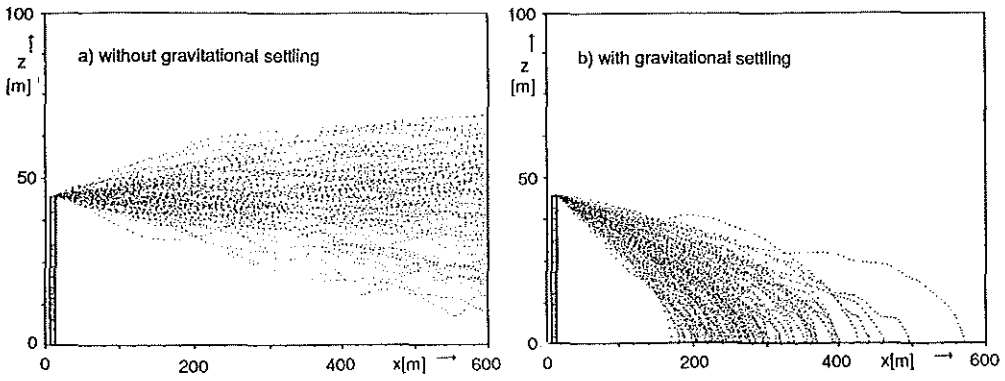

Figure 1: Influence of gravitational settling on atmospheric dispersion process (lateral view of 100 traces of LASAT-simulation particles, sedimentation velocity: 10 cm/s; turbulence according to diffusion category F)

Applying LASAT (with some 10000 simulated particles) the maximum of the ground-level concentration and of the deposited material has been found for stability category F at a shorter distance from source (200 m) than for the Gaussian plume model (1000 m). Corresponding to a shorter source distance higher maximum ground-level concentrations in air and on the ground have been found: The maximum value of dry deposition calculated with LASAT under stable conditions (stability class F) is

approximately a factor of 100 higher than the corresponding maximum value for the Gaussian plume model.

According to the procedure prescribed in the Incident Calculation Bases the dispersion conditions leading to the most unfavourable combination of ground-level concentration and deposition level have to be determined. The LASAT-calculations predict the most unfavourable deposition levels for dry deposition under very stable conditions (diffusion category F). On the other hand the Gaussian plume model predicts the highest deposition levels for wet deposition for the stable diffusion category E for which a rain intensity of 5 mm/h is assumed. This wet deposition level is approximately a factor of four smaller than the dry deposition level predicted by LASAT.

2.3 Dispersion and deposition of a poly-disperse aerosol

As a next step predictions of deposition levels have been compared for a broad size distribution of aerosol particles ranging up to 60 µm (poly-disperse aerosol). The LASAT-calulations indicate that dry deposition under very stable conditions (category F) leads to the highest deposition levels whereas for the Gaussian plume model wet deposition results in the highest values for the stable diffusion category E. But for the poly-disperse aerosol expected to be released in accident conditions the maximum deposition level calculated with LASAT is only by a factor of 2 greater than for the Gaussian plume model.

3. Discussion and conclusions

Differences have been identified in the predictions of environmental dispersion and deposition models for large particles between the Gaussian plume model and the particle simulation approach (LASAT) incorporating a simple boundary layer model. Especially for stable atmospheric diffusion (categories E and F) the differences are significant. Two reasons are responsible for this behaviour:

- The particle simulation approach considers particles with a sedimentation velocity in a more physical way than the Gaussian plume model. The continuous decline of the plume center line due to sedimentation as well as the corresponding shift of the maximum ground-level concentration towards the source can be described more realistically by particle simulation models. The Gaussian plume model does not reproduce this behaviour.
- The turbulence parameters used in the particle simulation model LASAT are based on similarity approach. The similarity approach needs boundary layer parameters like Monin-Obuchov-length L, mixing layer height h_{mix}, and roughness length z_0 as input parameters. Contrary to this the Gaussian plume model as described in the Incident Calculation Bases makes use of dispersion parameters which have been derived from dispersion experiments in a terrain characterized by a roughness length of about z_0 = 1.2 m. Concentration calculations performed with LASAT and the Gaussian plume model for particles without sedimentation show a fairly good agreement for unstable and neutral dispersion conditions. However, substantial differences have been found for stable conditions.

Based on the contemporary knowledge of modeling atmospheric turbulence the scientific community favours turbulence parametrization according to the similarity approach. This approach considers the detailed structure of the boundary layer (i.e. vertical profiles of turbulence) to a higher extent than other parametrization schemes and allows for considering continuously varying dispersion conditions. The approach uses approximations that are more universal than parametrizations like the potential law for dispersion parameters used by the straight-line Gaussian plume model. The performance of the similarity approach is considered to be high for slightly unstable, neutral, and slightly stable conditions. Research results on extremely unstable and especially stable diffusion conditions are rather sparse. Under these conditions the corresponding approximations in the similarity approach as well as in other turbulence parametrization schemes have the potential to deviate from real world observations [8], [9]. Due to this complexity a final decision whether the dispersion and deposition calculations with the

Gaussian plume model or with the particle simulation model LASAT are more reliable cannot be given here.

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AIR POLLUTION DISPERSION IN NONFLAT TERRAIN DURING NOCTURNAL CONDITIONS

M. Möllmann-Coers, F. Rohloff, E. Pomplun

Forschungszentrum Jülich GmbH, Abt. ASS-UW
Postfach 1913, D-5170 Jülich

Abstract: In the period between 1986 and 1990 the Research Centre Jülich carried out dispersion experiments in complex terrain. The case study presented here is based on a nocturnal experiment revealing a strong influence of a small hill on the dispersion in a stable stratified atmosphere. Whereas the air flow is calculated applying a nonhydrostatic flow model, three different types of model are used to simulate the dispersion, the Gaussian Model according the "Störfall-Leitlinien", a Segmented-Plume Model and a Lagrangian Particle Model. The results obtained by these different models are compared with measurements and are discussed with respect to the consequences in case of accidental releases of air pollution.

Introduction

Accidental releases of pollutants may cause severe impacts on the environment. An assessment of the contamination of the affected terrain with sufficient accuracy and in a reasonable time is a basic condition for effective and fast countermeasures. A great number of probabilistic computer models have been developed and are still under consideration for the simulation of pollutant transport by meteorological dispersion.

Up to now, the conventional procedure for calculating the concentration of pollutants in the environment subsequent to accidental releases has been based on the relatively simple Gaussian model. This model, however, is strongly limited insofar as it demands, inter alia, flat terrain as well as stable meteorological conditions. To circumvent these restrictions several modifications have been introduced. A consideration of the changes of the wind field in the Gaussian model has led to MUSEMET [*Straka*, 1981], a code able to make up-dates of the actual wind direction and velocity and in addition of new turbulence parameters at certain time steps.

From another class of models - quite different from the Gaussian ones with regard to their philosophy and their much greater complexity - the Lagrange-type dispersion model FITNAH IV [*Groß et al.*, 1986] may be used as a powerful tool in the successful simulation of a dispersion experiment in nonflat terrain. An application of this program for purposes of accidental release management, however, is not practicable due to its extensive CPU time.

In order to have an appropriate model for accident release conditions forecasting the pollutant concentration burden in the environment, we tried to adapt the MUSEMET code for nonflat terrain by incorporating wind fields which intrinsically consider the orographical characteristics of the terrain. Due to this improvement it was expected to find results which reflect the dispersion transport more realistically than by the stationary, flat terrain Gaussian model. To verify this assumption, both the Gaussian model and the modified MUSEMET code were applied here to the simulation of a tracer dispersion experiment. This experiment was the 34th in a large series performed in the late eighties by the Research Centre Jülich within the framework of an field experiments to study the influence of nonflat terrain on meteorological dispersion [*Zeuner and Heinemann*, 1990].

Furthermore, the FITNAH IV code will also be included in the comparison with the measured data in order to demonstrate the strength and weakness of the three approaches applied representing different degrees of complexity.

Experiment No. 34

From 1986 until 1990 dispersion experiments accompanied by extensive meteorological measurements were carried out at an overburden tip called "Sophienhöhe" near the Research Centre Jülich. This tip resulted during development of an open-pit mine. One of the experiments, no. 34, revealing a strong influence of the tip on the stable stratified airflow (see Fig. 1), was chosen for this investigation. The experiment was carried out in the evening hours of September 19, 1989. Tracer sampling took place between 9 p.m. and 9.30 p.m., that is about two hours after sunset and two hours after the beginning of emission. A strong inversion had built up with a temperature gradient of 2.1 °C/100m, measured at the meteorological tower of the Research Centre where the emission source was fixed 50m above the surface. The wind measured there was 223 degrees as indicated by an arrow in Fig. 1. The concentration pattern is characterized by high values close to the emission point. Maximum concentration was found at a source distance of about 310 m in wind direction. A rather high concentration was also found at the slope of the tip tilted

towards the emission point and also at the western and southern foot of the tip. However, a very low concentration was found at the top of the tip.

Models

To simulate this experiment three types of dispersion model were used. Two of these models need a proper wind field. Therefore a flow model was applied to precalculate the airflow around the tip.

The airflow was calculated using the nonhydrostatic flow model FITNAH described in detail for example by *Groß* [1985, 1986, 1987, 1988, 1989]. FITNAH makes use of the equations of motion, the first law of thermodynamics, and the conservation equation of humidity, which are integrated numerically. In addition, an elliptic differential equation for the pressure deviation, with the condition of anelasticity, is solved numerically using a method given by *Chorin* [1962]. The turbulent fluxes are parameterized by a 1.5 order closure using a prognostic equation for the turbulence energy [*Bischoff-Gauß*, 1982] and *Blackadar's* [1962] mixing length hypothesis.

The following three dispersion models were applied in this study:

(1) the well known Gaussian Model was used without modifications concerning roughness or nonflat terrain. To perform a standard calculation the dispersion parameters were taken from the "Störfall-Leitlinien" [*BMU*, 1989]. These parameters were evaluated from dispersion experiments performed in the seventies at KFA Jülich and KfK Karlsruhe. Due to the temperature gradient and wind speed at the meteorological tower the dispersion parameters for diffusion category F related to *Pasquill*'s classification scheme were used here.

(2) the MUSEMET code, which is of a Segmented-Plume Model type. It is formulated as Volume-Source Model, i.e. a Gaussian dispersion is determined for a specific time step, yielding the tracer concentration at each point of the 3D space. Then a new Gaussian dispersion starts for the next time step using these concentration values as a new source together with actual data for wind speed and wind direction as well as new parameters considering atmospheric stability. An analytical integration over all these Gaussian processes yields a new formula of the Gaussian type. Therefore suitable dispersion parameters are also required. In the investigation presented here the parameters according to *Islitzer* [1963] based on vector wind vane measurements are applied to be in better agreement with reality than a standard calculation with the simpler standard Gaussian model. Using MUSEMET it is possible to follow changing weather conditions during the diffusion process. Nonflat terrain conditions are considered implicitly by also changing, in addition to wind speed and wind direction, the effective release height along the streamline that starts at the emission point. More details of this type of model are discussed by *Straka* et al. [1981].

(3) the FITNAH IV model, which in contrast to the flow model FITNAH is a dispersion model of the Lagrangian type. It is based on a simple Monte Carlo scheme to perform transport and diffusion by following particles along their path through the turbulent airflow time step by time step beginning at the emission point. The position of each particle at one moment is calculated by starting at its position at the preceding time step. The mean flow calculated from the FITNAH flow model at this position is superposed by a turbulent component derived from the Lagrangian timescale using a Gaussian distributed random number generator and considering the particles' motion one time step before. The resulting effective wind vector is multiplied by the time step to yield the position of the actual time step. The Lagrangian time scale is a function of the turbulent kinetic energy calculated from the FITNAH flow model. The time steps are chosen as a quarter of the actual Lagrangian time scale. Details are given elsewhere [*Groß* et al., 1986].

Results

The result of the Gaussian model is shown in Fig. 2. The concentration bars are scaled as in Fig. 1. A comparison between measured and calculated concentration values is given as scatter plot (Fig. 5). All values are divided by the maximum measured value. Fig. 2 and also Fig. 5 show that especially higher concentrations are strongly underestimated, although about 50% of the calculated values do not differ by more than a factor of 10 as shown in Fig. 6

Fig. 3 shows the results of the MUSEMET code. Here the calculated wind field has been taken into account. Because of the property of the Volume Source Model following the airflow, it leads to higher concentration values at the right flank of "Sophienhöhe" due to the streamline that starts at the emission point and passes over the southern flank down to the pit. In combination with applying dispersion parameters based on direct turbulence measurements, higher concentrations are somewhat better represented than in the standard Gaussion calculation as depicted in Fig. 3 and Fig. 5. The worse

representation of the whole concentration pattern compared with the Gaussian calculation (see Fig. 6) is at least partly an artificial effect since it was not possible to take air samples down in the pit where better results might be expected due to the possibility of MUSEMET considering the change in wind speed and wind direction along the path trough the wind field. So the worse representation of the measurements by the simple Gaussian model here could not be counterbalanced by possibly better results from MUSEMET.

The results of the FITNAH IV Lagrangian model is shown in Fig. 4. Like in the MUSEMET model the mean concentration follows the streamline started at the emission point. But a part of the plume moves over the western flank of the tip and as found in the tracer experiment, a very low concentration is calculated at the top of "Sophienhöhe". But contrast to the experiment, a very low concentration was calculated at the stagnation point of the airflow at the tip. The overall representation of the concentration pattern is given in Fig. 6. More than 75% of the values are better than a factor of 10 and more than 50% better than a factor of five. 20% are within a factor of two. Although maximum concentrations are underestimated by more than a factor of two, the results are significantly better than using the Gaussian models (see Fig. 5).

Conclusion

Three different models attempted to simulate a tracer experiment in nonflat terrain under nocturnal conditions: (a) the Gaussian model as is envisaged by the "Störfall-Leitlinien" [BMU, 1983], (b) the Gaussian-based modified MUSEMET code which - in contrast to model (a) - considers the changes of wind direction and velocity as well as nonflat terrain conditions by varying the effective emission height, and (c) the nonhydrostatic flow model FITNAH in combination with the Lagrangian dispersion model FITNAH IV [Groß et al., 1986].

The Gaussian model gives the concentration distribution as expected from the basic assumptions of this model. Since it cannot recognize any change in the wind direction - which is forced here by the hill - it cannot follow the stream lines around the hill either as proven to appear in the experiment. The restriction to apply this model to nonflat terrain only is clearly demonstrated here.

The results from the modified MUSEMET code show significant deviations from the experimental findings, moreover, a basic problem of tracing plumes in nonflat terrain by Gaussian-based codes becomes obvious. The splitting of the plume by the influence of the "Sophienhöhe" - one of the major results of the experiment considered here - cannot be simulated by Gaussian models in general.

Not surprisingly, the FITNAH code yields by far the most realistic data of all three models. Both the position and the concentration values are in rather good agreement with the experiment.

Summing up, it may be said that first of all for a sufficiently realistic prognosis of air pollution impact in nonflat terrain a propper wind field is indispensable. Secondly, even if a proper wind field is available, the use of a Gaussian plume-type model leads to unsatisfactory results, especially in situations with plume split effects like the one presented here. Using a Lagrangian particle model is a promising method if it is possible to run such a model in a CPU time of a few minutes.

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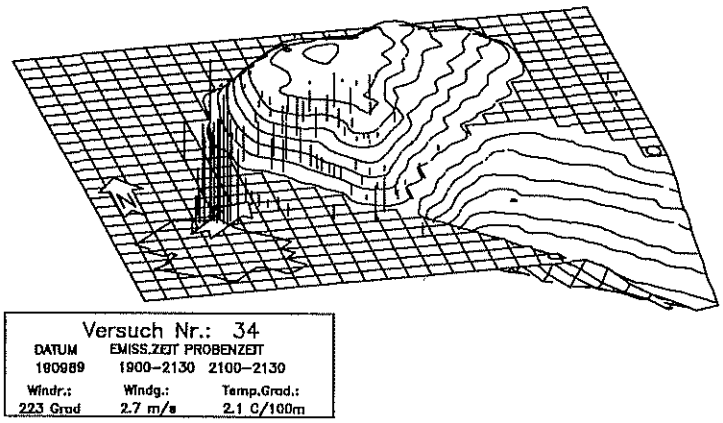


Fig. 1: Results of dispersion experiment no 34. The site of the Research Centre is indicated by a polygon and the north direction is given by a north arrow. The tip and the pit are represented by isolines at a vertical distance of 25 m. The grid spaceing is 250 m. The thick vertical bar marks the emission point. Small vertical bars show the mean concentration of the tracer at the respective position related to the half-hour sampling time. Small circles indicate that there was no tracer detected above the atmospheric background concentration.

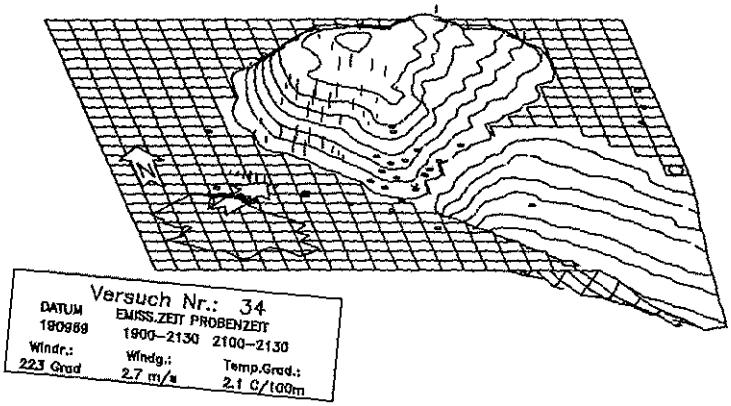


Fig. 2: Results of Gaussian model. The vertical bars representing the concentration are scaled as in Fig.1.

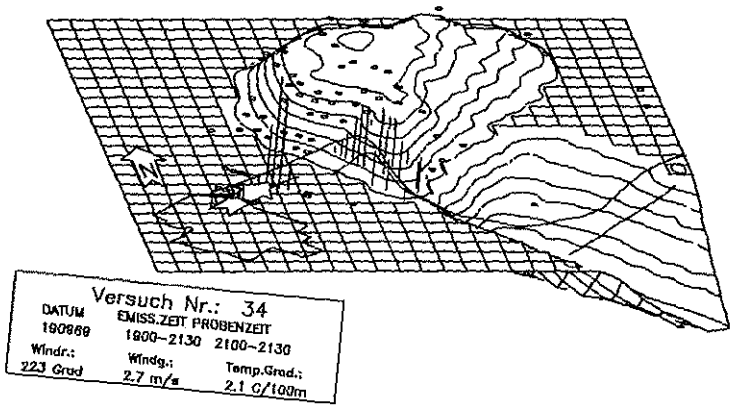


Fig. 3: Results of MUSEMET code. The vertical bars representing the concentration are scaled as in Fig. 1. In addition the streamline starting at the source and its vertical projection to the surface is outlined.

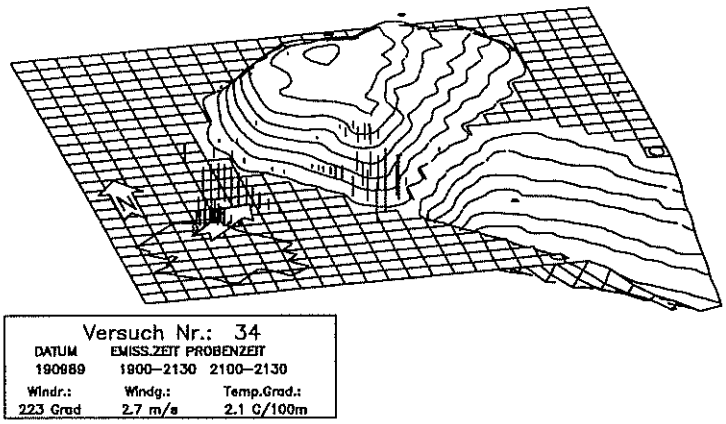


Fig. 4: Results of FITNAH IV model. The vertical bars representing the concentration are scaled as in Fig. 1.

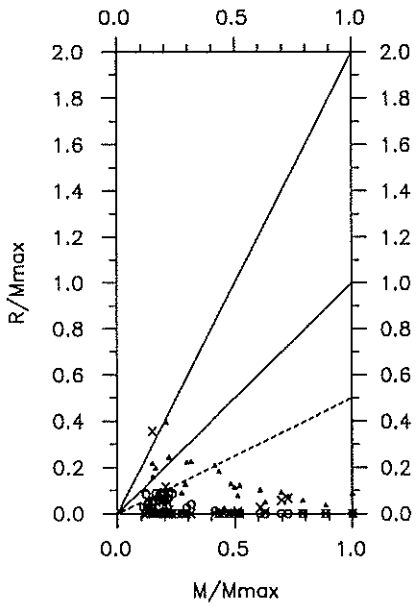


Fig. 5: The results of the three dispersion models are plotted against the measured values, Gaussian: circles, MUSEMET: crosses, FITHNAH IV: triangles. All values are normalized to the maximum measured value. The dashed lines enclose the area in which the relation of measured and calculated result is less than a factor of two. Measured concentration values less than 5% of the maximum value are omitted to avoid an unmethodical cluster of points near the origin.

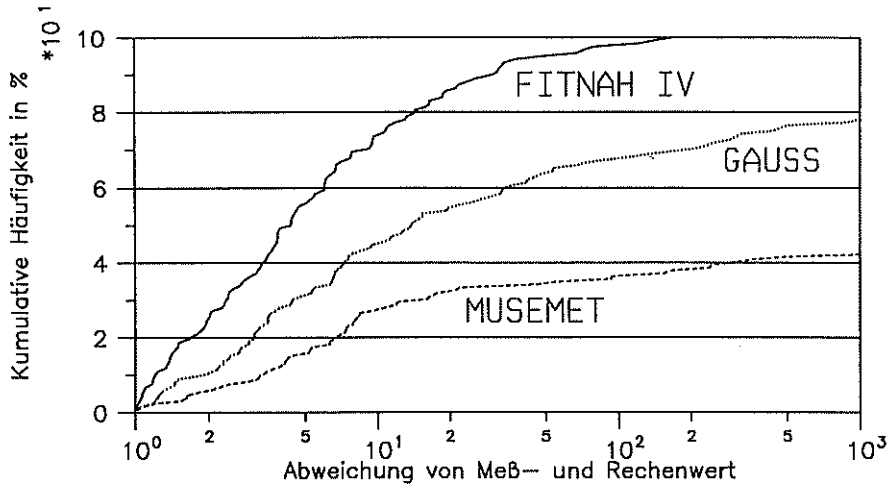


Fig. 6: Cumulative frequency distribution of the three dispersion models results for all sampling positions. The percentage of those values closer to the measured ones than a factor given on the abscissa can be read from the ordinate. This graph, however, gives no information on how well the measured maximum values are represented by the model.

DETECTION OF LOCAL AEREAS WITH MAIN DOSE IMPACT FOR EMERGENCY PLANNING IN SURROUNDINGS OF NUCLEAR POWER STATIONS

Dieter Kaspar
Umweltministerium Baden-Württemberg, Kernerplatz 9, D-7000 Sturrgart 1

Dr. G. Hehn
Institut für Kernenergetik und Energiesysteme
Universität Stuttgart, Pfaffenwaldring 31, D-7000 Stuttgart 80

Summary

The detection of a nuclear accident and its follow up are one of the main tasks of the remote sensing system (KFÜ). For dose calculations in the surroundings normally analytical Gaussian solutions are used, which are not very reliable in complexe terrain. To get improvements, numerical procedures can be applied which need the use of supercomputers. Taking the nuclear power stations of Neckarwestheim (GKN) and Obrigheim (KWO) for example thecapacity is shown, and results are presented. Depending of the stack hight the spread of the plume shows different shape: At KWO the plume stays mainly in the river valley, at GKN the surrounding hills are more affected.

1. The remote sensing system in Baden-Württemberg (KFÜ)

The KFÜ in Baden - Württemberg is now working since 1982. It is an equipment of the supervising authority and enables it to permanently check important parameters of the nuclear power plants at Obrigheim, Neckarwestheim and Philippsburg during normal operation as well as in an emergency.
The main tasks are :

- supervising the nominal power operation of the nuclear power stations
- detecting irregularities
- trigger immediate automatic alarm at the authorities if boundary values are exeeded
- support the radiation expert in the emergency staff giving him
- immediate estimation and prognosis of the radiation dose in the environment during an emergency considering the actual situation of release and the weather conditions.

In order to meet these demands in each of the five nucclear power stations two computers are installed which are connected with many devices to obtain for instance data of:
- the release of radioactivity to the air
- the release of radioactivity to the water
- the dose rate in the environment
- several internal parameters as there are neutron flux, pressure and temperature in special regions of the reactor.
- the weather conditions

The measured and processed data are available to the ministry of the environment a well as for the boards for emergency protection and for the operators of the nuclea power plants. Fig .1 shows the installations of the KFÜ.

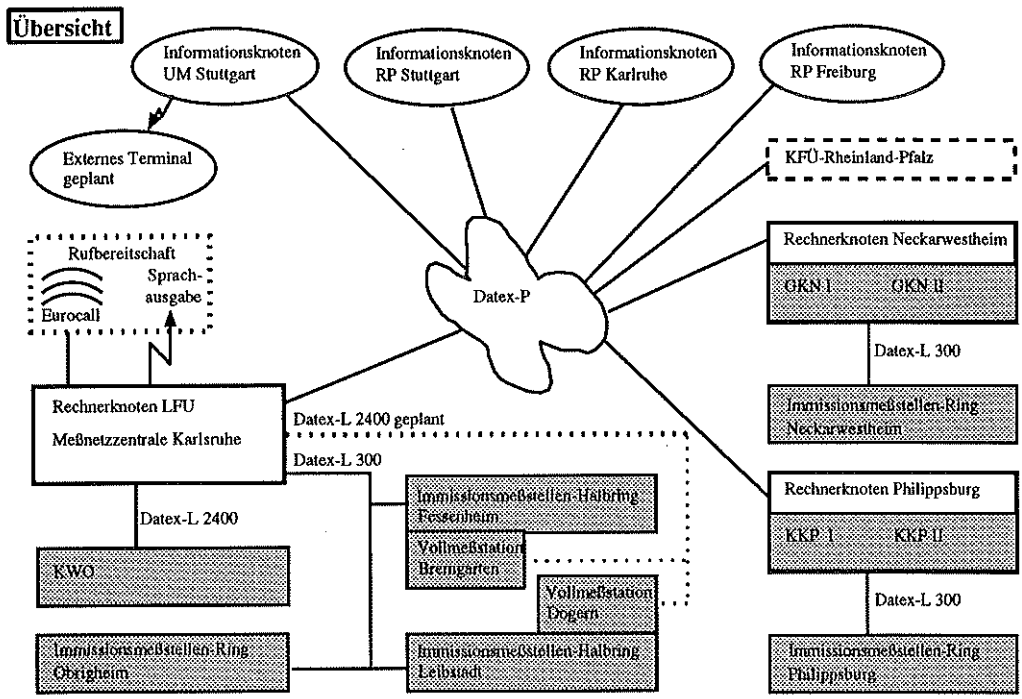
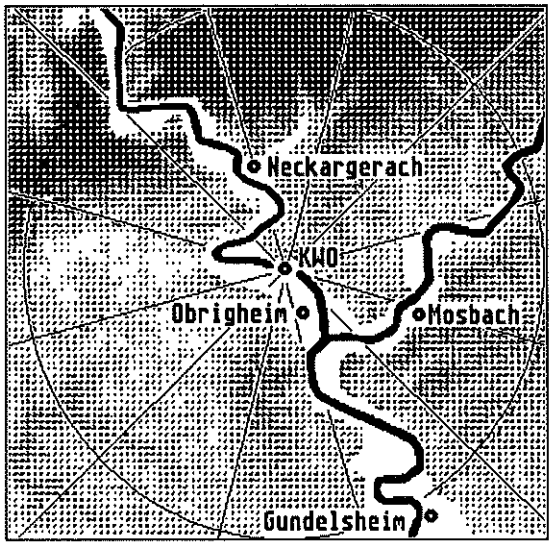


Fig. 1: Kernreaktor-Fernüberwachung Baden-Württemberg

Using the results of measuring the release of radioactivity to the air and the weather conditions e. g. speed and velocity of the wind, the system calculates automatically or on demand the radiation dosage in the surroundings. The results can be presented as isodose lines. For these calculation a simple Gaussian code is used. It is therefore not possible to consider the complexity of the environment. To improve this shortcoming a new code has been applied, making use of the processing capacity of the supercomputer CRAY 2 of the university of Stuttgart.

2. The SPEEDI code for calculation the radiation dosage

Speedi is a mass consistent model for wind fields over complexe terrain for the real-time assesment of environmental consequences due to accidental releaseof radioactivity. It has been developed in the USA. Later on it has been improved in Japan and has been installed at the Cray 2 for the special terrain in the environment of Obrigheim (Fig. 2) and Neckarwestheim with deep valley and many hills.



Topography of the local site (20 x 20 km) of KWO (Obrigheim)
with the applied wind sectors, ground height ranging from 110 to 540 m.

Fig. 2 The topography of the surroundings of Obrigheim

SPEEDI consists of three main parts:
1. a mass consistent wind flow model for terrain induced airflow
2. a particle-in-cell model for athmospheric dispersion
3. a method for concentrations and dose calculations

The code takes the results of the weather and release measurements directly from the KFÜ and calculates the concentration of radioactivity and the doses in the surroundings.

The results show a great improvement compared with the calculations of the Gaussian model: at KWO, where the stack doesn't exceed the valley, the plume follows the main valley and some smaller valleys,there are more higher affected aereas far from the plant. The KFÜ results based on the Gaussian model show a very straight spread of the plume with the maximum dose value closer to the stack (Fig. 3 and 4). Fig. 5 and 6 show the SPEEDI and Gauss results of concentration calculations in the surroundings of Neckarwestheim. In this case SPEEDI calculates more aereas of great concentrations, one of which is very close to the plant, the others are farer away on slopes and hilltops. The maximum of concentration from the Gauss calculation is located about 3 km off the plant far in the valley of the river Neckar. The reason for this different behaviour compared with KWO is the 160 m stack, which surpasses the edge of the valley by 50 m. The results can bee seen on the KFÜ terminals.
The radiation expert of the emergency staff can compare the calculated results with measured dose rates from many stationary measuring equipments. Thus he is well prepared for advising the staff leader inn respect of carrying out emergency measures.

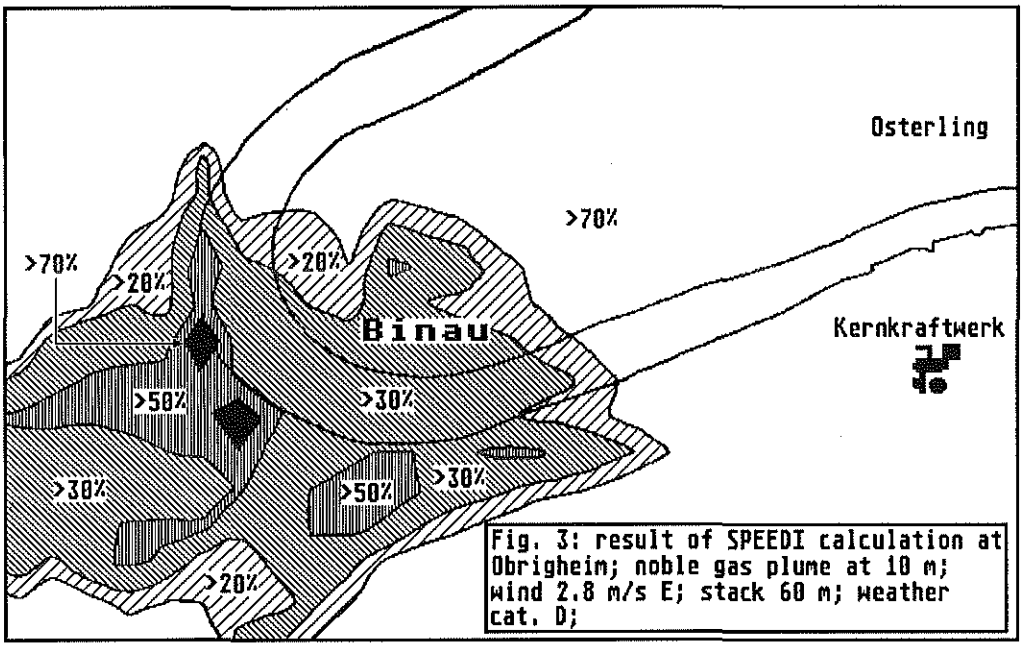


Fig. 3: result of SPEEDI calculation at Obrigheim; noble gas plume at 10 m; wind 2.8 m/s E; stack 60 m; weather cat. D;

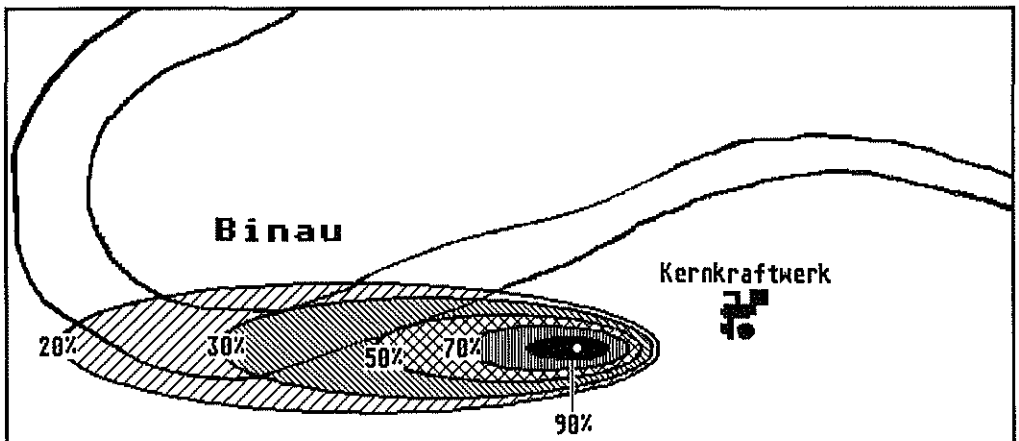


Fig. 4: Results of KFÜ calculations, same conditions as Fig. 3.

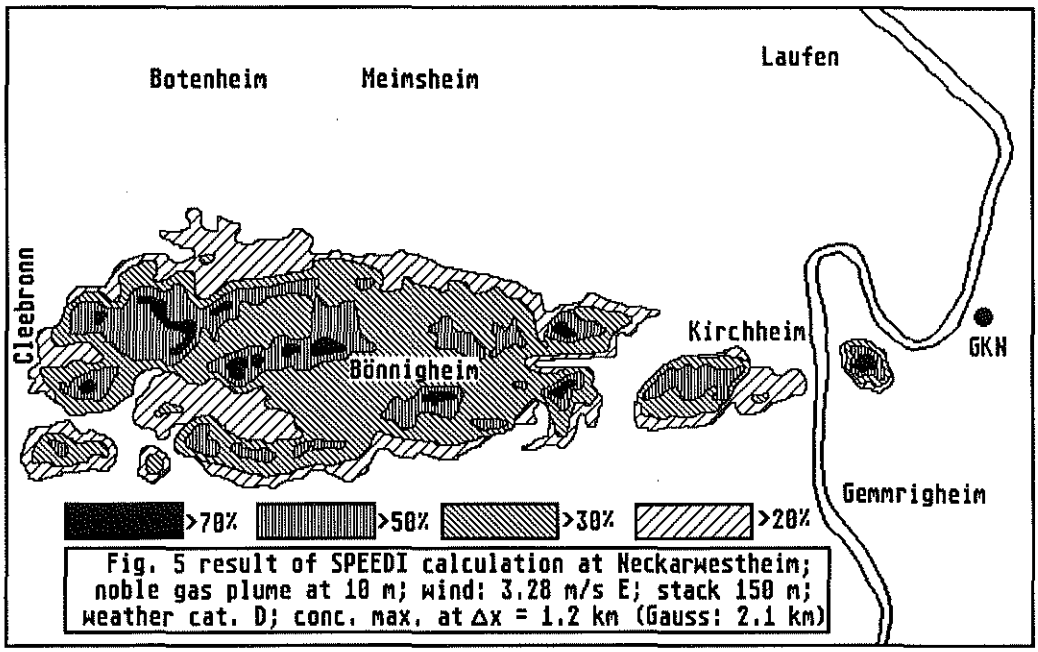


Fig. 5 result of SPEEDI calculation at Neckarwestheim; noble gas plume at 10 m; wind: 3.28 m/s E; stack 150 m; weather cat. D; conc. max. at Δx = 1.2 km (Gauss: 2.1 km)

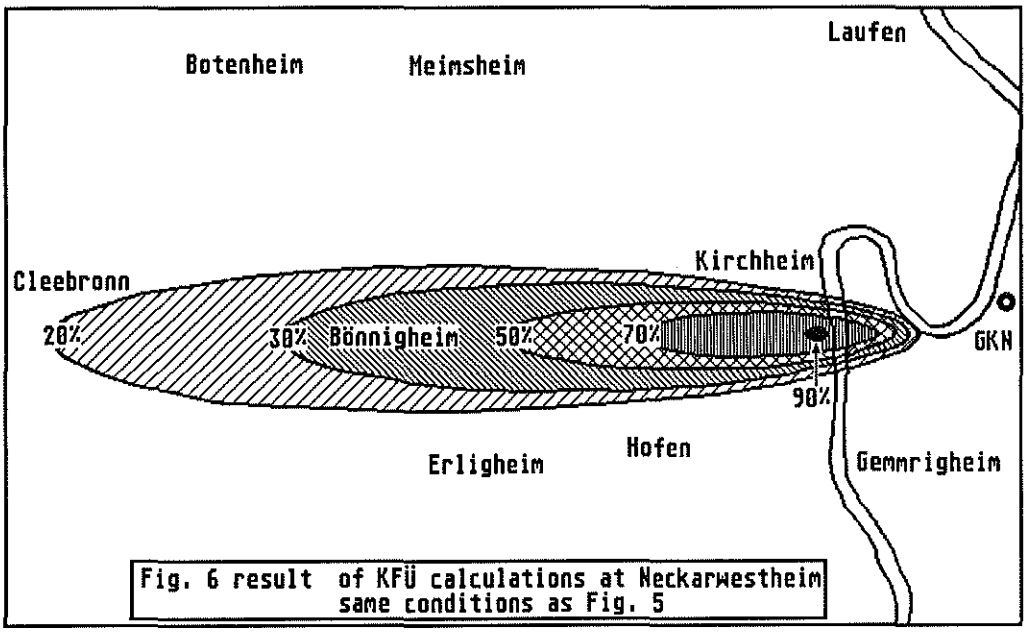


Fig. 6 result of KFÜ calculations at Neckarwestheim same conditions as Fig. 5

CALCULATION OF THE DISPERSION OF RADIONUCLIDES IN FLOWING WATERS USING A DYNAMIC MODEL

H. Höfer[1]
Universität Karlsruhe
Institut für Kerntechnik und
Reaktorsicherheit
Postfach 6980
D-7500 Karlsruhe

A. Bayer
Bundesamt für Strahlenschutz
Institut für Strahlenhygiene
Postfach 11 08
D-8042 Neuherberg

Summary

A one-dimensional model is presented by means of which the distribution of radionuclides can be calculated over the environmental sectors water, suspended matter, and sediment along a river. Corresponding to the three interacting environmental sectors, the model is based on three coupled differential equations which have to be solved numerically. In the example given, a Runge-Kutta-method was used. There are presently limitations on the validation of the model because corresponding measurements are not yet available. By way of examples, a calculation of four hours' release of Cs-137 into the Upper Rhine is made.

1. Introduction

The dynamic models for radionuclide dispersion in flowing waters can be subdivided into two groups. The first comprises models by means of which the radionuclide distribution is calculated exclusively within the water body, i.e. interactions of radionuclides released are not accounted for; a series of such models is given in [1].

The second group of dynamic models to calculate the dispersion of radionuclides in flowing waters differs from the first by additionally describing the activity concentration distribution of each radionuclide considered over the environmental sectors of interest, i.e. water, suspended matter and sediment. These models require a great number of parameters, especially to describe the transfer from one environmental sector to another.

2. Dynamic model to calculate the dispersion of radionuclides

The model described in the following is based on the static model KIRMES [2]. It is used to calculate the distribution over the environmental sectors water, suspended matter and sediment of radioactive substances released into a flowing water as a function of time and site. For this purpose the river is subdivided into defined longitudinal sections according to its flow direction. The model describes both the interaction of the three environmental sectors within a river section and the interlinkage of these sections. This is shown in Fig. 1.

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[1] Seit 01.10.92 bei ABB Reaktor GmbH, Mannheim

The radionuclides are assumed to immediately mix homogenously within each river section. Mathematically, the model represents a system of 3 coupled, ordinary differential equations of the 1st order.

Allowance is made for radionuclide and river specific data which can be varied for each individual section in principle, i.e. adapted to the real conditions.

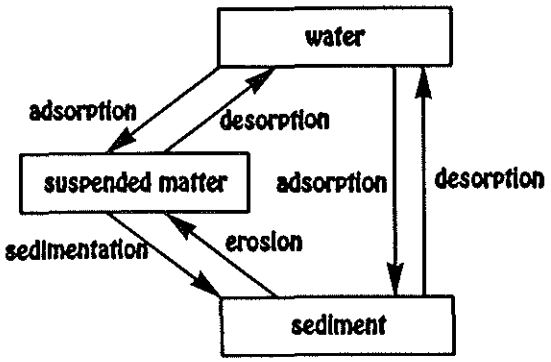

Figure 1a: Linkage of environmental sectors within a river section

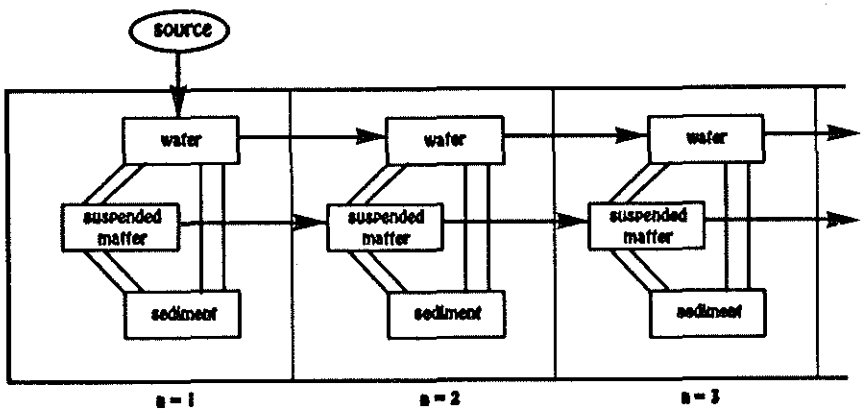

Figure 1b: Linkage of environmental sectors within the river sections and connection of river sections

3. Validation of the model

The validation of the model was performed in [3] based on measurement of the Co-60 content of suspended matter in the river Weser due to releases from the nuclear power plant Würgassen and conducted at two different measurement sites [4]. The available quarterly mean values of Co-60-release were taken as source term.

The model is particularly suited for calculations after short-term releases. The time needed to establish the state of equilibrium in reference to suspended matter

amounts to about 30h. Thus the source term assumed to be constant over a period of 3 months (ca. 2000h) must be regarded as a long-term source term. The measured values are also available as quarterly mean values. The dynamic behaviour of the model could thus not as yet be validated.

In addition to the above mentioned shortcomings of the source-term, there were only values available for validation at two measurement sites, defining a flow segment greater than 110 km. This illustrates that only a limited validation could be performed, i.e. of the linkage of water and suspended matter for a long-term source term. There is satisfactory agreement between the available measurements and the results calculated using the model.

4. Example of a calculation by means of the present model

The source term assumed in the following example amounts to a total of $3.7*10^{10}$ Bq Cs-137 released into the outlet channel and homogeneously distributed over a period of 4 hours. The considered reference site is the Upper Rhine (e.g. near Biblis).

For the short-term source term, 200 river kilometers from the site of release were considered up to a time of about 40 h after occurence of the contaminated wave. The results from these calculations are shown in Figures 2-4.

The absolute maximum values of radionuclide concentrations in the different environmental sectors were found to occur at different times and, thus, at different sites. The maximum value in water is reached within the first section, where the release occurs; it amounts to about $2.3*10^{3}$ Bqm^{-3}. As a result of dilution and the process of adsorption, the concentration within the water decreases over the further course of the river.

The complete adsorption of radionuclides by suspended matter is a process requiring a certain time. Suspended matter, which is subject to motion reaches its maximum value of activity concentration about 18 h after the onset of release, 60 km below the site of release, amounting to $9.6*10^{3}$ Bq kg^{-1}, based on the selected assumptions and parameters. During the further course, it slowly decreases because the dilution (dispersion) by non-contaminated suspended matter always causes a process contrary to adsorption so that the theoretical equilibrium characterized by the K_d-value is never reached.

The sediment in this model is linked to the water primarily as a result of the adsorption interaction, and the maximum value of activity concentration in the sediments is therefore also reached within the first section, amounting to $6*10^{1}$ Bq kg^{-1} (referred to dry mass). Adsorption to sediment is a process still slower than that to suspended matter (transfer rate smaller by a factor of 10 [5]), and, additionally, the active sediment layer is not subject to motion in the model, so that the adsorption comes to an end when the radioactive wave has passed. Comparing the transportation velocity of sediment of about 3 km a^{-1} with the velocities of water and suspended matter of about 1,2 ms^{-1}, it is obviously justified to neglect the sediment transport.

The radionuclide concentration of sediments remains approximately at the value reached following the passage of the radioactively contaminated wave of water and suspended matter. The concentrations in the environmental sectors water and suspended matter, too, turn to a largely constant after-run at a low level after a certain time. Using the model, this may be explained by the fact that the three environmental sectors are permanently interlinked by interactions between adsorption and desorption. After passage of the contaminated wave the sediment, assumed to be static in the present context, is storing acitivity which returns radionuclides to the water via the process of desorption.

5. Limitations of the model

Viewing the complexity of the actual procedures of dispersion in nature, the results from calculations conducted by means of the present model are only of limited relevance. In the following, some examples are given to illustrate the conditions which are not accounted for by the model.

The model represents the river as a rectangular straight channel. Although depth, width and other parameters may be varied by section, the influence of bends, the natural form of the river bed, slack water regions (ports, groyne fields, etc.), hydro power plants, shipping and others, remains unconsidered.

Referring to the radionuclides, there are also some effects still unconsidered, e.g. competitive ions, such as K^+ or complexation by complexing agents such as nitrilotriacetic acid. Grain-size effects of the adsorbent but also the content of suspended matter (adsorbent concentration) will influence transfer rates and distribution coefficients. In this field, therefore, many correlations must be clarified and accounted for in the model in order to enable the natural processes to be more precisely represented.

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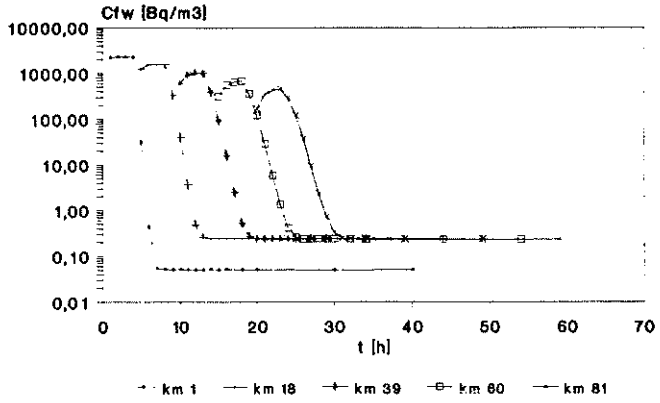

Fig. 2: Calculated activity concentration in water as a function of time and distance

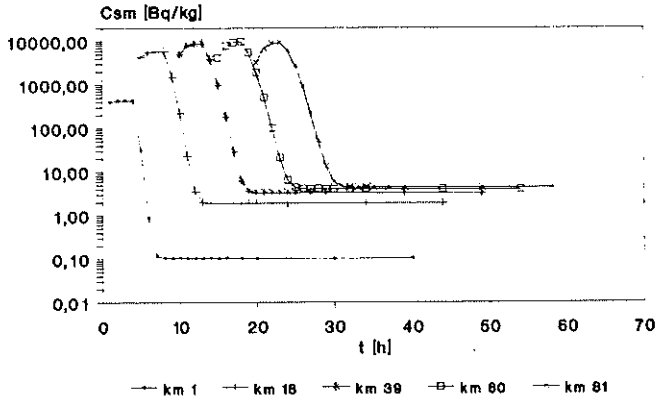

Fig. 3: Calculated activity concentration in suspended matter as a function of time and distance

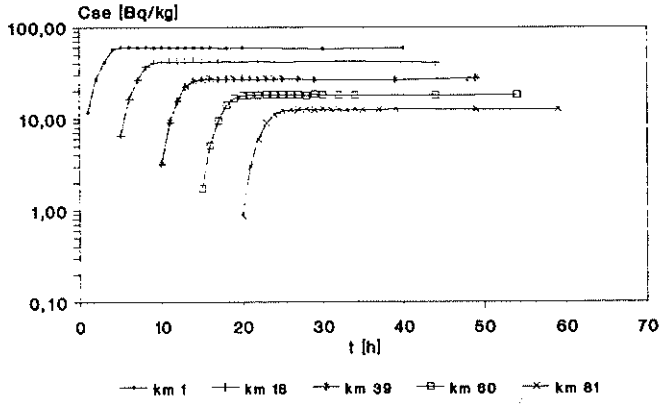

Fig. 4: Calculated activity concentration in sediment as a function of time and distance

APPLICATION OF THE *AVV UNDER Sec. 45 StrSchV* IN PRACTICING THE ATOMIC ENERGY LAW

Anna Maria Rall,
Technical Inspection Agency of Bavaria and Saxony, Munich, FRG

Horst Wolf,
Ministery for Economy, Middle Classes and Technology of the Federal State (Land) North Rhine-Westphalia, Düsseldorf, FRG

Summary

Since 1989 an amended version of the Radiological Protection Ordinance "Strahlenschutzverordnung" (StrlSchV) [1] is in force in the Federal Republic of Germany. According to it a strongly formalized procedure (AVV 45 [2]) for evaluating the radiation exposure is layed down for the examination whether the technical layout and the operation of a nuclear facility are planned in a way that the exposure of humans due to emission of radioactive materials does not exceed the dose limits fixed as the effective dose and as the partial body doses. An example will be shown that this procedure presents a sufficient base for official decisions even in the presence of disturbing influences on the dispersion (e.g. complex orography, building structures, thermally caused circulation systems). It is necessary however according to AVV 45 to investigate the dispersion conditions experimentally (e.g. by use of a wind tunnel) and theoretically (e.g. by use of suitable flow and dispersion models).

1. The significance of the AVV under Sec. 45 StrlSchV in practicing the atomic energy law

In the course of the operation of nuclear facilities there usually occur discharges of radioactive materials into the environment. This is one of the reasons why the construction and operation of nuclear facilities are subject to official supervision (licencing duty and official surveillance) based on laws (in Germany the Atomic Energy Act and the Radiological Protection Ordinance is decisive). Official control is executed in atomic energy licensing and surveillance procedures. Here the applicant or the operating company respectively has to prove among other facts that the technical layout and the operation of the plant fulfil the requirements of the Sec. 45 (1) StrlSchV [1]. According to it the radiation exposure of humans that is caused by the discharge of radioactive materials from nuclear facilities by exhaust air and liquid waste must not exceed posted limits of body doses per calender year (dose limits) e.g. 300 μSv for the effective dose. The examination whether the dose limits of the Sec. 45 (1) StrlSchV [1] are adhered to occurs according to Sec. 45 (2) StrlSchV [1] for the reference person (adult, infant) at the most unfavourable point of impact taking into account the exposure pathways, the way of life of the reference person and other assumptions layed down in the StrlSchV [1], which must be valued in the calculation of the radiation exposure. The single items of the combination of all substantial parameters in a model for the calculation of the doses are regulated based on the side constraints named in the StrlSchV in the general administrative regulation "Allgemeinen Verwaltungsvorschrift under Sec. 45 StrlSchV: evaluation of the radiation exposure resulting from radioactive effluents from nuclear plants and facilities" (AVV 45) [2]. The importan-

ce of the AVV 45 for nuclear legal procedures is based on the fact that the dose limits fixed in the AVV 45 may be regarded as being adhered to if the models and parameters of the AVV 45 are used for the evaluation of the radiation exposure by the discharge of radioactive materials from nuclear plants and facilities.

2. The model for dose calculation of the AVV 45, especially the model of the dispersion of radioactive materials in the atmosphere

By use of the model for dose calculation of the AVV 45 [2] those points in the surrounding area of nuclear facilities are evaluated where the highest radiation exposure of the reference person ("most unfavourable point of impact" [1, annex 1]) must be anticipated caused by the distribution of discharged radioactivity in the media of the environment taking into account real and possible future uses like being present there or by consuming food produced there. Insofar the distribution of the emissions are caused by atmospheric dispersion processes their quantitative determination occurs by a dispersion calculation which is based on the so called Gaussian plume model and on a meteorological statistic specific to the site which has been calculated from data that cover 5 years at least. According to this model the concentration distribution of a continually emitting source averaged over time shows a Gaussian normal distribution horizontally as well as vertically. These procedures and the required parameters for calculation are documented in AVV 45 [2]; the model parameters that have to be used are valid for homogeneous and stationary dispersion conditions and for elevated point sources. The requirement of homogenity and stationarity of the flow comprises an ideal view of the natural conditions which is not met at many sites in reality. Important requirements of the Gaussian plume model are not met if for example the distance to the emitting source is very small or very large or if a low emission heigth is given or if the surrounding landscape is not sufficiently flat (angle of inclination $> 5°$) or if the dispersion of the exhaust is disturbed by marked conditions specific to the site (e.g. cooling towers, building- or landscape structures). In such cases the quantitative calculation of the site specific atmospheric dispersion conditions according to AVV 45 [2, chapter 4.6] can only be carried out based on experimental tracer gas investigations on a model true to scale in a wind tunnel and/or based on the use of other procedures [1, annex XI, III.3, 2. sentence] like special flow simulations (e.g. model "FITNAH") and dispersion calculations (e.g. model "LAGRANGE"). Fig. 1 shows the stack heigths of nuclear power plants using light-water reactors within the area within which the "Atomic energy act [3]" is operative including the heigths of surrounding elevations and cooling towers. The figure shows that there exist cases in practicing the atomic energy law for which the use of the Gaussian plume model does not reach the goal without additional detailed investigations.

3. Experiences in the application of the AVV 45: an example:

From the cases shown in Fig. 1 the site specific dispersion conditions for the case KKW-O have been investigated after the AVV 45 (1990) has come into force and have been compared to a reference case KKW-R. The case KKW-O is described by a complex orographic site (roughly structured landscape, site in a valley with side valleys and a curve with special channelling effects and a steep slope very close to the plant) and caused by that a formation of thermal circulation systems (autochthone wind systems) and additionally characterized by a low stack heigth. The reference case KKW-R is described by a flat landscape; the stack heigth has not been changed as compared to the

case of KKW-O. As an additional variable a so called synthetic diffusion statistic has been considered for the comparison of KKW-O/KKW-R (according to AVV 45 [2, chapter 4.5.2, 3. paragraph, 2. dash] this is possible in principle but it is not the preferred alternative) and a diffusion statistic based on meteorological data measured on site (this is the preferred alternative according to AVV 45 [2, chapter 4.5.2, 1. and 2. paragraph).

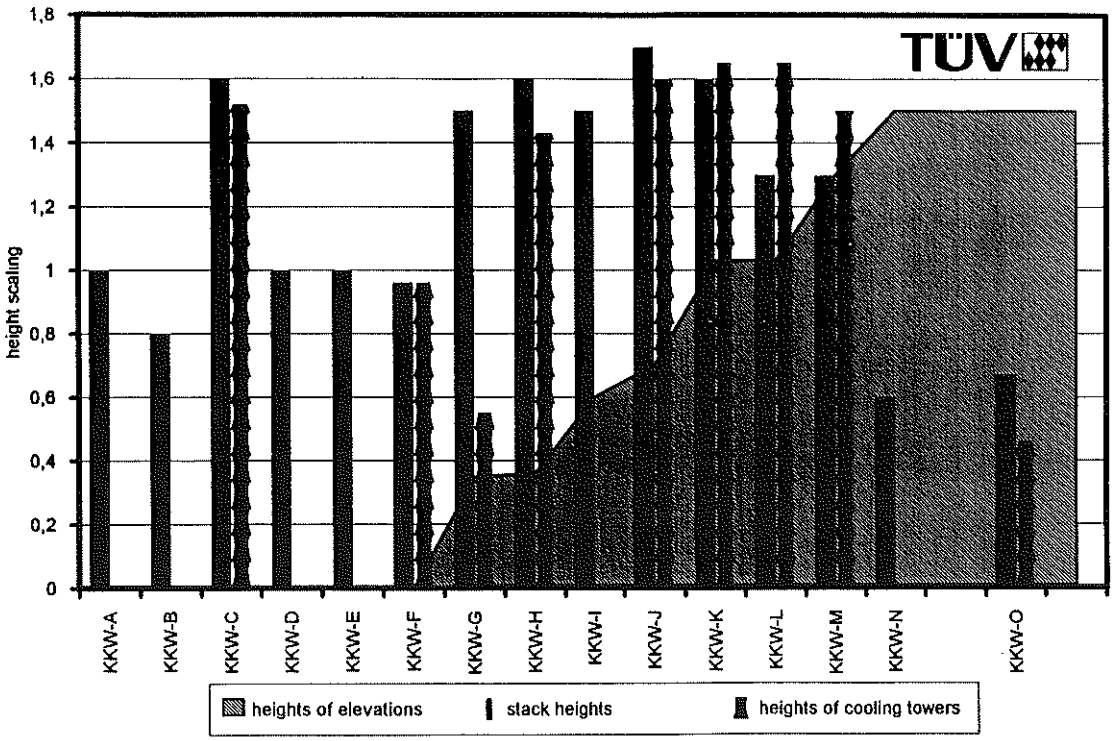

Fig. 1: Comparison of stack heigths of nuclear power plants using light-water reactors within the area within which the "Atomic energy act [3]" is operative including the heigths of surrounding elevations and cooling towers

3.1 The influence of site specific mechanically caused aerodynamical effects on the concentration distribution of radioactive materials

As a result of the wind tunnel experiments for KKW-O spatial dispersion factors v are obtained as well as modified model parameters (emission heigths H_{eff} depending on sector and distance and Gaussian dispersion parameters Σ_y and Σ_z) along the trajectories. Based on them it is possible to carry out the calculation of the dispersion according to AVV 45 [2] for the case KKW-O. The calculation of the dispersion for the reference case KKW-R was carried out according to AVV 45 using the parameter sets given there. The difference relevant to the dispersion and concentration distribution in the sector containing the most unfavourable point of impact becomes especially clear (Fig. 2) by the comparison of the normalized long term dispersion factors (NLDF) depending on distance, as a measure for the resulting concentration distribution near to ground in the close vicinity

of the emittent for the cases of KKW-O and KKW-R. From the comparison the conclusion can be drawn that in the case of site specific dispersion conditions the resulting concentration distribution in the close vicinity of the emittent can only be calculated with respect to the goal of protection (that is without any underestimation) if number one a site specific dispersion statistic is used and number two in accordance with AVV 45 [2] detailed investigations of the site specific mechanically caused aerodynamical effects are conducted, e.g. by use of wind tunnel experiments.

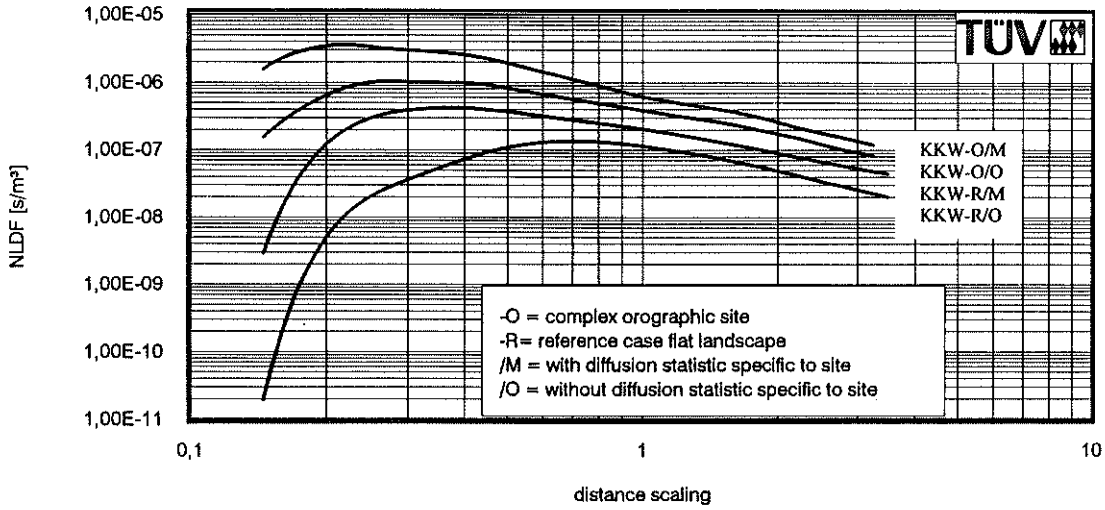

Fig. 2: Normalized long term dispersion factors (NLDF) in the sector of the most unfavourable point of impact [1, annex I]

3.2 The influence of thermally caused aerodynmical effects on the concentration distribution of radioactive materials

If complex site specific dispersion situations are present which are influenced by the formation of thermally caused circulation systems (autochthone wind systems) their influence can be quantitatively evaluated by the use of mesoscalic flow simulations (e.g. model "FITNAH") in combination with LAGRANGE-type dispersion calculations. Fig. 3 shows the distribution of the NLDF obtained by this procedure under taking into account the autochthone wind systems. Their contribution to the NLDF has been as a result of LAGRANGE-type dispersion calculations weighted according to their probability and superposed on the NLDF obtained from calculations based on the Gaussian plume model. The example shows that the NLDF caused by autochthone wind systems can reach about the same order of magnitude as the NLDF calculated by use of the Gaussian plume model with the use of modified model parameters (case KKW-O: sole consideration of mechanically caused aerodynamical effects). Additionally it is to mention that the concentration maxima may lie at different points (in the case of the example there are two maxima with a distance from each other of a few hundred meters). In the case of complex orografic sites significantly higher effective NLDF may occur in the course of the superposition of the contributions from the calculations based on the Gaussian plume model and from the calculation of the dispersion for autochthone wind systems. All in all the consideration of the influence of autochthone wind systems on the NLDFs shows that in

the case of special site specific dispersion conditions the resulting concentration distributions can only be calculated with respect to the goal of protection (that is without any underestimation) if in accordance with the special case regulations of StrlSchV and AVV 45 (compare 2.) detailed investigations of the site specific thermally caused aerodynamical effects are rigorously conducted e.g. by numerical simulations with the use of flow and dispersion models.

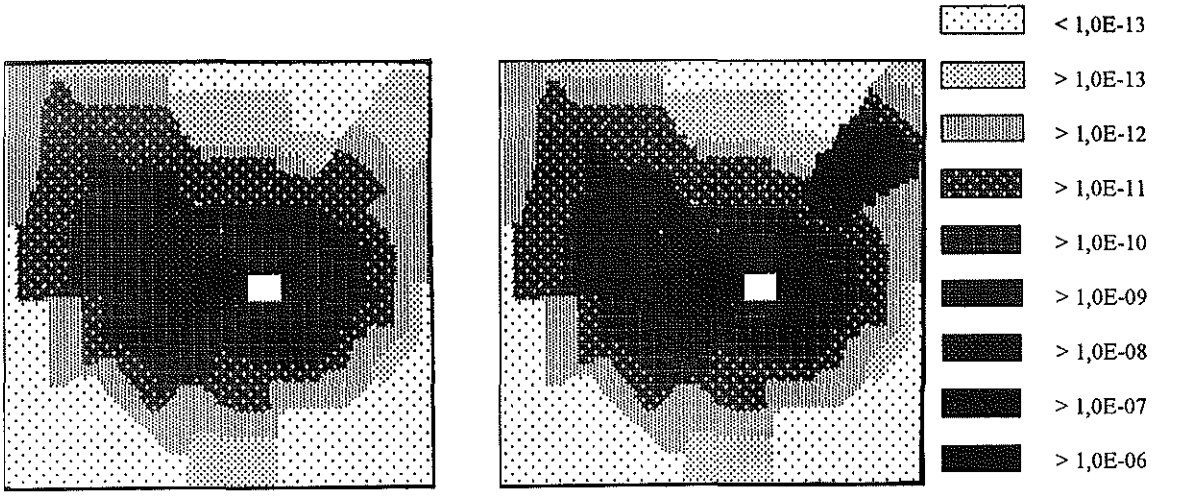

Fig. 3: Comparison of the distribution of the normalized long term dispersion factors (NLDF)
a) neglecting
b) taking into account
the autochthone wind systems

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MODELLING THE BIOSPHERE FOR RADIOACTIVE WASTE REPOSITORIES

F van Dorp† and R A Klos‡
† NAGRA, Wettingen, 5430 Wettingen, Switzerland.
‡ Paul Scherrer Institute, Würenlingen, 5223 Villigen PSI, Switzerland.

Summary.

Modelling the transport of radionuclides in the biosphere and the estimation of the potential radiological exposures that arise from their accumulation in environmental media are important parts of safety assessments for repository concepts. This paper describes the philosophy of model construction, and demonstrates the application of the newly developed *Terrestrial - Aquatic Model of the Environment*. New features in *TAME* include aquatic systems, solid material transport and the rigorous application of mass balance to transport processes.

1. Aim of Modelling for Waste Disposal Assessments.

Calculations of the potential consequences of the transport and accumulation of radionuclides in the human environment following their release from radioactive waste repositories are used to provide estimates of the performance of radioactive waste repository systems. The multi-barrier approach adopted in Switzerland seeks to ensure that any return of radionuclides to the human environment, should it occur, will take place after the radioactive decay of the most active radionuclides and in such low concentrations as to present an insignificant health risk to members of the public. The effect on the health of members of exposed groups can best be estimated by using the concept of radiological dose[♦], which is a measure of the health effects arising from the exposure to radionuclides. Use of the concept of dose is advantageous since it provides a means of comparing the consequences of different radionuclides on an equal basis.

Results from biosphere models for waste disposal should **not** be seen as predictions of doses that will be received by members of exposed populations in future times. The return of radioactivity to the human environment takes place over long timescales (in excess of 1000 years) and so the state of the climate, the nature of the biosphere and the form of human civilisation cannot be estimated with precision. Results should therefore be seen as an expression of the magnitude of the doses likely to arise given the set of modelling assumptions. Careful selection and justification of the modelling framework are therefore required to ensure that a full and representative set of features, events and processes is included.

In estimating the consequences the aim is to **not underestimate** the magnitudes of the potential doses but equally importantly, unrealistically pessimistic assumptions should also be avoided since they can lead not only to overestimates of the potential doses but also to models that are unable to distinguish between different situations. This has meant that the degree of detail in biosphere models for performance assessments has increased in recent years, since it is not always easy to see whether a given assumption is pessimistic or not because of the complex feedback and retention mechanisms and parameter interdependencies in the biosphere.

2. Model Structure.

The most commonly used tool in biosphere modelling for radioactive waste disposal is the donor-controlled linear first-order compartment model in which dN_i/dt, the rate of change of the contents of each part of the biosphere (represented by a distinct *compartment* - e.g. a river or soil) is determined by the sum of terms for direct external input, radioactive decay and ingrowth and for the transfers of contaminants between compartments. These transfers are represented by *transfer coefficients*, λ_{ij}, which are defined as the fractional transfer rates of the contents (N_i) from compartment i to compartment j:

♦ The term *dose* is used here to mean the effective dose equivalent and is the sum of the weighted dose equivalents in specific organs from the intake of activity into the body in one year, plus the sum of weighted dose equivalents from external irradiation in one year. This definition corresponds to the ICRP 26[1] definition of dose.

$$\lambda_{ij} = \frac{1}{N_i}\frac{dN_j}{dt} \tag{1}$$

This type of model[2] has evolved over many years as a result of field observations in radioecology and well-established methods exist for solving the transport equation; for example the BIOPATH[3] code package is used in Switzerland for waste disposal assessments.

The role of the biosphere modeller, therefore, is to find ways of adequately representing the physical, chemical and biological processes that together determine the coefficient for each potential transfer in the system. In principle, all that is required to determine the compartmental inventories as a function of time are the values for the transfer coefficients, λ_{ij}, and one way of obtaining these would be via field measurements for a given site. In practice, however, because the λ_{ij}s can only be inferred from the measured environmental concentrations of tracers, such a site investigation programme could not yield a database sufficiently broad as to be able to represent the conditions at the site over the long timescales for which the assessment must be carried out. Furthermore the transport coefficients are not uniquely defined by single processes, for example, the transport of contaminants to the rooting soil zone (top soil) from the deeper soils can include advection, bioturbation, diffusion and erosion, so that the transfer coefficient is given by a linear sum of terms

$$\lambda_{DT} = \lambda_{DT}^{advection} + \lambda_{DT}^{bioturbation} + \lambda_{DT}^{diffusion} + \lambda_{DT}^{erosion} \tag{2}$$

A single value (or range of values) based on site measurements cannot satisfactorily represent the contributions from these different processes. The disadvantages of working with the transfer coefficients themselves as basic inputs to the models were illustrated in the first phase of the international biosphere model validation study, BIOMOVS[4], where, in the early stages, the modelling of transport was carried out using values for the transfer coefficients alone. As the exercise progressed it became increasingly clear that different participants included different processes in the transfer coefficients and this made the task of understanding the results of the modelling exercise extremely difficult.

The most effective way of estimating the transfer coefficients is therefore to use the more fundamental (and more readily measurable) site characteristics. A generic system for the definition of transfer coefficients based on site specific data then allows the uncertainty in the model parameters to be investigated in a systematic manner (for example in the PSACOIN Level 1b code intercomparison[5]), yielding information about which are the most sensitive parameters affecting the system. This approach to the identification of the transfer coefficients has been pursued in the development of the *Terrestrial - Aquatic Model of the Environment (TAME)*[6]. This new model is being used for the biosphere modelling in the current Swiss assessments for the disposal options for low- and intermediate level wastes and for high level waste. In *TAME*, the transfer coefficient between the deep soil and the top soil is characterised in terms of 10 parameters representing the physical, chemical and biological conditions in the deep soil (and a corresponding number for the top soil). All solute (i.e. water) fluxes and solid material fluxes are also integrated in a system of mass balance between the compartments.

In contrast to the radionuclide transport in the biosphere, modelling the doses received by inhabitants of the biosphere is based on the equilibrium values in the food chain and other environmental media. This is valid because the timescales of interest in waste disposal assessments are much longer than the dynamics of the accumulation in, for example, milk or meat, which proceed on timescales of days. This is one major difference between biosphere modelling for waste disposal assessments and for accident consequence assessments, where the timescale of interest could be of the order of days. In models such as *TAME* the dose from the exposure pathway p (D_p) uses the dose per unit exposure D_{exp} (i.e. by ingestion, inhalation, external irradiation), the associated exposure rate (E_p) and processing factor ($P_{p,i}$) for the inventory N_i via pathway p so that the dose from this pathway is

$$D_p = \sum_{i,exp} P_{p,i} E_p D_{exp} N_i \tag{3}$$

The processing factors can be complex, involving many factors, though computationally straightforward. For example the consumption of dairy products requires parameters relating to the amount of contaminated water a dairy animal consumes (from different sources). Pasture consumed by livestock may have become contaminated

by root uptake from the top soil as well as from the irrigation of the pasture by well water or surface water. Intake of soil by the animal during grazing is also accounted for. In practice the exposure factors *Ep* depend on the desired end-point of the calculation. These can be annual individual dose, to a maximally exposed individual or group of individuals (the *critical group* concept) or to representative individuals or groups that, by virtue of estimations of *average* behavioural habits can be seen as typical members of the exposed population. The end point of the *TAME* calculations is annual individual dose because the legal framework for radiological protection with regard to waste disposal facilities in Switzerland is written in terms of this quantity.

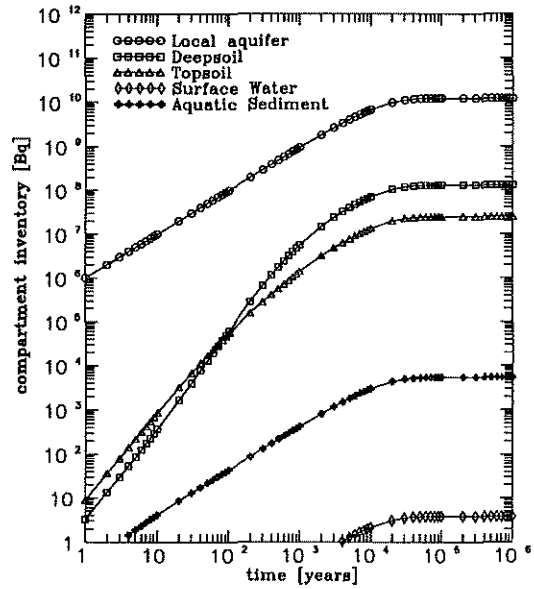
3. Example Calculations.

In waste disposal assessments the source term to the biosphere depends in a detailed way on the performance of the waste material and the repository structures as well as on the details of the groundwater transport in the geology around the repository. The form of the source term chosen for these example calculations is deliberately chosen so as to make no assumptions about these other important aspects of the performance of the overall disposal concept. The results discussed here are not representative of the performance of waste disposal concepts under investigation by NAGRA and are only an illustration of the functioning of biosphere models for waste disposal.

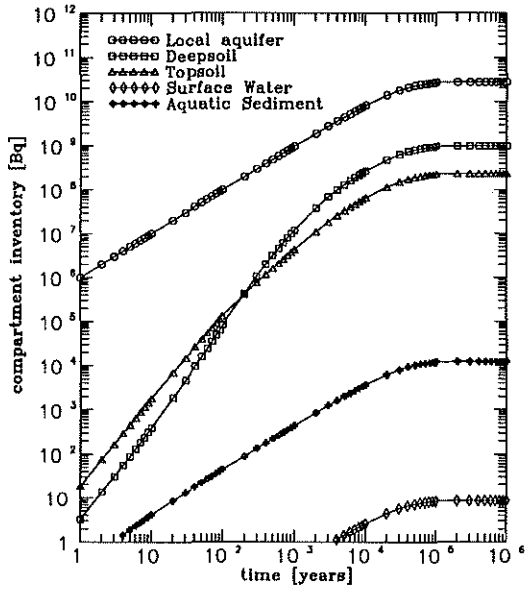
The release of 10^6 Bq y^{-1} for 10^6 y of ^{135}Cs is assumed to the near surface aquifer (*Local Aquifer* in the nomenclature of *TAME*) of a small valley side biosphere that is representative of those found in central Switzerland in the present day. ^{135}Cs is an important waste product from fission reactors and it has been chosen because of its relatively highly sorbing chemistry and its long halflife of 2.3 10^6 y.

TAME representations of the biosphere identify four other compartments in addition to the *Local Aquifer*. The *Top Soil* is the rooting zone of soils for pasture and crops. The *Deep Soil* lies between the *Top Soil* and the *Local Aquifer*. The *Surface Water* in this biosphere representation is a pond that has formed at the outcropping of the *Local Aquifer* and discharges out of the system. *Aquatic Sediments* are deposited in the pond as a result of the accumulation of soils in the pond by erosion. The transport processes between these compartments in the model include rainfall, evapotranspiration, bioturbation, diffusion, capillary rise, erosion and dredging. The pond is used as a source of drinking water for both animals and the human population of the release region as well as for the irrigation of pasture land and crops. Similarly a well in the aquifer could also be used for these purposes. Doses are calculated on the basis of an individual obtaining the full annual dietary requirement from the agricultural produce of the contaminated region. The food pathways are drinking water, milk and dairy products, meat, vegetables and fish obtained from the pond.

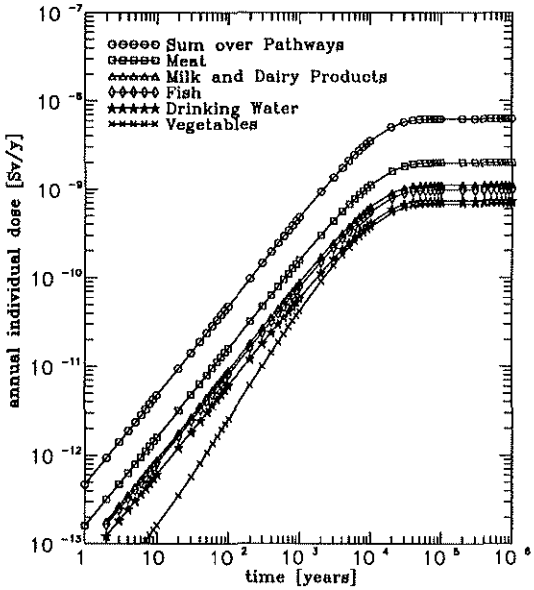
A variation in the parameters defining the climate is considered here. The *Wet Climate* has a net annual precipitation excess of 2 m y^{-1}, and the *Dry Climate* that has a precipitation excess of only 0.1 m y^{-1}, necessitating the use of the well in the local aquifer for irrigation of crops and pasture. It is assumed that the aquifer is not directly influenced by climate. The results for the compartment inventories as a function of time (Figure 1(a) and 1(b)) illustrate the differences in the transport model of these two biosphere representations. In general the compartment contents are higher in the dry climate. This is a result of the relatively lower water fluxes through the soil and hence greater overall retention. Furthermore the use of the *Local Aquifer* for irrigation purposes in the dry climate means that there is a significantly greater transfer to the *Top Soil* compartment than in the wet climate and this, combined with the greater water flux from the *Top Soil* to the *Deep Soil* as a result of higher infiltrating rainfall in the wet climate gives an order of magnitude higher concentration in the *Top Soil* in the dry climate. Figures 1(c) and 1(d) show the differences in the annual individual doses between these two climate states. Despite the order of magnitude increase in the *Top soil* concentration, the total dose increases in the dry climate only by a factor of around five because the *Top soil* is not the only compartment involved in the exposure of individuals. The change in the dose from vegetable consumption illustrates the processes involved since in the wet climate D_{veg} is the least important and in the dry climate it is the second most important. The rates of change of the doses suggest that it is the water concentrations that have the most influence on the overall dose for most of the pathways. The rate of change of D_{veg} increases markedly as the root uptake mechanism in vegetables begins to have a significant effect on the vegetable consumption dose.



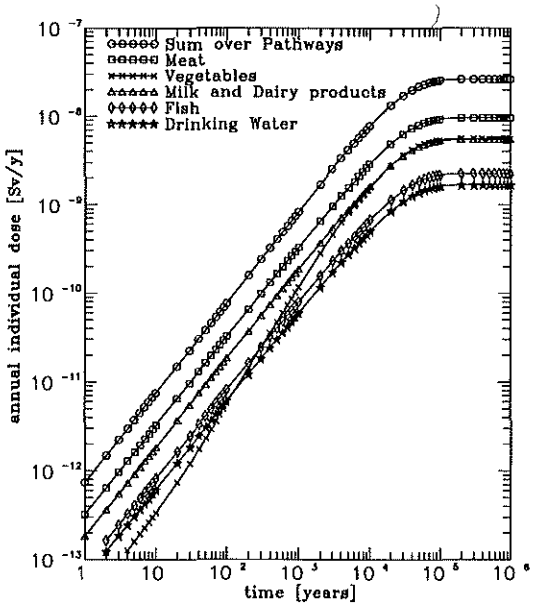
(a) Compartment Inventories: Wet climate state.



(b) Compartment Inventories: Dry climate state.



(c) Annual Individual Doses: Wet climate state.



(d) Annual Individual Doses: Dry climate state.

Figure 1 – Example results for the release of ^{135}Cs to a valley side biosphere in central Switzerland. The time evolution of compartment inventories and annual individual doses are illustrated in a climate state wetter than the present day and in a climate state drier than the present day and in which greater use is made of local water resources.

4. Conclusions.

The above example illustrates how the *Terrestrial - Aquatic Model of the Environment* can be used to distinguish between modelling situations on the basis of the site characteristic parameters. The calculations here focus on the representation of different biosphere states but the flexibility of the model is such that both local sensitivity analyses (where single parameters or groups of parameters are allowed to vary in a deterministic manner) and global sensitivity analyses (where input parameters are allowed to vary simultaneously and stochastically) can be used to identify the most important model parameters contributing to the radiological consequences of the release to the biosphere and which parameters contribute most to the overall uncertainty in the estimation of the doses received. This approach allows the modeller to dispense with the broad pessimistic assumptions necessary in less detailed models of the biosphere.

TAME features many improvements compared with previous models[7,8,9] of the biosphere used for waste disposal assessments. These include

- representation of the aquatic environment, which includes rives, ponds and lakes;
- solid material transport;
- mass balance of water and solid material fluxes as site specific input;

The aquatic environment is important because release from the geosphere is most likely to be to the aquatic environment and because accumulation downstream from the release site (in river deltas for example) can be accounted for.

There remain a number of important points in the field of biosphere modelling for waste disposal assessments. The second phase of BIOMOVS is now underway and this will provide the opportunity for the continued intercomparison of models and datasets in use internationally. In particular the case specification for the *Complementary Studies* exercise[10] is designed to test individual modelling aspects of waste disposal models. Together with the *Reference Biospheres* task group[11] the aim is to establish a consensus on methodology and databases. The methodology for modelling changes in biosphere characteristics of the long timescales in waste disposal assessments also needs to be formalised.

Input data for models such as *TAME* can be classified in two ways: as a model database or as an experimental results database. The methodology for establishing model databases is not well defined with the consequence that suitable model database values are difficult to obtain with accuracy and the data values used in the assessments must therefore have an associated large uncertainty (which can also be difficult to quantify).

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MESOSCALE OCEANIC EDDIES AND DISPERSION IN THE DEEP WATER AT THE DUMPSITE FOR LOW LEVEL RADIOACTIVE WASTE IN THE N-E ATLANTIC

François NYFFELER et Eduard ZUUR

Group PROSPER - University of Neuchâtel -11 Rue Emile Argand
CH-2007 NEUCHÂTEL

Summary

A three dimensional general circulation model is used to simulate the dispersion of dissolved matter from a bottom source, in the near field around the dumpsite for low level radioactive waste in the north-east Atlantic. The aim of the study is to investigate the combined effect of benthic storms and topography on the vertical transport in the deep ocean. Typical eddies are spun up numerically by applying a tangential stress at the upper limit of the model in order to generate a velocity field comparable with the observations carried out at the site. The results show that a tracer homogeneously distributed in the benthic mixed layer is washed out and dispersed in the overlying water in a time scale of days.

1. Introduction

Low level radioactive waste dumping operations have been carried out in the north-east Atlantic since 1974 up to 1982 under the control of the **N**uclear **E**nergy **A**gency (NEA/OECD). NEA is also in charge to carry out periodically site suitability reviews, taking into account the updated scientific knowledge. The radiological impact resulting from dumping is calculated in terms of collective and/or individual doses. The assessments take into account the type and the quantities of radionuclides, as well as the critical pathways by which they may return to man. In the last site suitability review, a compartment model was used to quantify the water flows in the ocean [1]. Several scenarios of release from the drums have been examined and even with conservative hypothesis, the calculated doses to the population resulting from the past dumping operations remain several order of magnitude below the ICRP limits (ICRP= International **C**ommittee for **R**adio **P**rotection). They represent in any case a minor fraction of the doses of natural or fall-out origin [2]. However, recognising the importance of scientific continuity and the critical role of reliable models, NEA set out to establish a co-ordinated research programme to increase the knowledge's of the processes controlling radionuclide transports to biota and humans [3]. In particular, measurements carried out in the vicinity of the site during the NOAMP experiment (**N**ord**Ö**st **A**tlantisches **M**onitoring **P**rogram) revealed the existence of mesoscale eddies and of energetic benthic storms [4]. Hundreds of nephelometric profiles obtained during the NOAMP programme in the surrounding of the dumpsite also confirm that suspended particulate matter is present up to several hundreds meters above the homogeneous bottom mixed layer while the shape of the profiles indicate a bottom origin [5].

Although it is not established yet to which extent the eddies in the surface layer may contribute to triggering benthic storms, bottom reaching eddies have often been observed in energetic areas of the ocean, and some kind of coupling can reasonably be postulated. Forcing of the deep flow by the steep bottom topography may also play a key role in the dispersion by accelerating the bottom current, thus promoting local resupension and eventually enhancing the vertical transport. As a matter of fact, the ocean box -model used in the last assessment is not suited to resolve such mesoscale processes and features which may be of relevance to the environmental impact on marine organisms living in the near field .

2. The Prosper General Circulation Model (PGCM)

The general circulation model developed by the PROSPER group was applied to simulate concentration fields generated by typical benthic storms, in a topographical situation corresponding to the dumpsite in the Northeast Atlantic (46°N/17°W) . The PGCM is a three dimensional model based on the Navier-Stokes equations, an equation of state and the heat equation. The model uses the rigid lid and hydrostatic approximations [6]. The area covered by the simulations is 155 km x 114 km, with an horizontal grid mesh of 6 km. The water column is subdivided in 25 layers, between 3000 m and 4700 m. Orlanski conditions are prescribed at the open boundaries of the domain [7].

A benthic storm can be roughly defined as a transient state, during which the velocities near the bottom exceed 10 cm/s for a duration of more than 2-3 days. Because ocean dynamics are inherently unstable, we endeavoured more to illuminate the processes leading to dispersion than trying to reproduce any situation corresponding to a specific set of observations. However, in order to make the simulations realistic, typical dynamical features (size of the eddies, horizontal velocity, temperature gradient) were extracted from the whole set of observations carried out during the NOAMP programme, and were used as input to the model.

It is well known that fluid instabilities predominates in the generation of eddies. However, in our application the eddies are spun up numerically inside the domain by applying a tangential stress at the upper limit of the model taken here at a depth of 3000m. In fact, as we are mainly interested in the dynamics of the lower part of the water column, we do not need to resolve the surface Eckman layer nor the energy transfer in the upper part of the ocean. The forcing is thus merely a convenient mean to generate an eddy field in the deep ocean which is coherent with the length scale and velocities reported in the NOAMP area adjoining the dumpsite [4]. The radius adopted for the eddy (35 km) is close to the Rossby radius, and the stress is adjusted to obtain velocities O(10cm/s) at 1500m above the bottom.

2.2 The simulations

The simulations summarised below correspond mainly to the case of instantaneous releases of radionuclides, one of the most critical situation considered in the last site suitability review. As the time scale covered by these simulations is of the order of weeks, the natural decay of activity is negligible for most nuclides compared to the effect of dispersion. The tracer is thus considered to be stable although a decay could be readily included in the formulation for further simulations, based on the embedding of our regional model into a basin scale, eddy resolving, general circulation model.

Three typical dynamic simulations are considered for comparison:

a) A cyclonic eddy above a flat bottom
b) The same eddy above the dumpsite topography.
c) An anticyclonic eddy above the dumpsite topography.

In each case, the eddy is spun up for a period of 10 days in order to generate a consistent flow field. As the model does not resolve the small scale near bottom processes, the tracer is considered to be homogeneously dissolved in the last compartment over the bottom, the thickness of which corresponds to the benthic mixed layer (50m). The drums are supposed to be uniformly distributed over the central valley of the site. The geometry of the valley is included in the flat bottom case in order to use the same source term for the dispersion study. Two scenarios are applied for releasing the tracer from the drums. The first scenario is a true instantaneous release in the bottom layer whereas a continuous and constant source is considered in the second case. Because of the dynamical nature of the processes, a graphic animation package is used to analyse and to compare the evolution of the concentration fields.

2.3. Results

The undulations of the isopycnals for case b) and case c) are quite similar to those observed well above the topographical features during the oceanographic campaigns, increasing thus the confidence in the model. The comparison of the horizontal and vertical flow charts with the flat bottom case shows that, while the horizontal velocities are only weakly effected by the topographic forcing, the steep bottom topography results in a significant increase of the vertical velocities. The dispersion towards the interior of the ocean occurs in a few days, but the thermal stratification is not affected by the diapycnal transport of the tracer.

3. Conclusions

Horizontal advection and diffusion along isopycnal surfaces are dominant in the ocean and have been used as input parameters to drive the ocean scale dispersion models in the hazards assessment for dumped radioactive waste. Recent oceanographic studies raise the question of the role of the eddy dynamics in the oceanic circulation and mixing processes, taking into account the respective horizontal and vertical dimensions of the ocean. The work presented here is a first attempt to assess the vertical transfer from a bottom source under the influence of a single typical eddy in the region of the actual dumpsite. More detailed statistics on the characteristics of the eddies in that region (e.g. frequency, size, life time) would be needed to generalise the concept. These preliminary results suggest that the regional vertical transfer induced by individual mesoscale eddies may be significant and can hardly be accounted for by the implicit diffusivity coefficient assumed in the coarse compartment models, in which there is much averaging over space and time scales.

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CONTINUOUS PROPAGATION COMPUTATION FOR RADIOACTIVE EMISSIONS USING SUITABLE PC SOFTWARE

Authors:

Author: Dipl. Ingenieur **Stephan Pawlytsch**
Co-Author Dipl. Physiker **Erich Neuburger**

Company:

Admintec GmbH
Bayernstraße 53
W-7530 Pforzheim
Germany

Phone: (0) 7231 / 38 08 65
Fax: (0) 7231 / 38 08 57

Summary

The Program ENVIROf or computing the dose resulting from gamma submersion, ground radiation and inhalation after emission of radionuclides via chimneys.

ENVIRO can be integrated directly in the Radiation-Protection ATLAS. A graphic representation for all measuring points and computations is implemented.

1. Field of application

Enviro is a PC program for use by e.g. the adviser of those in charge of disaster relief in the field of radiation protection.

It is intended to help him to quickly and reliably determine the dose (dose rate) resulting at specific locations after emission of radioactivity from the following types of exposure:

- External radiation exposure from the radioactive cloud (gamma submersion)
- Radiation exposure from inhaled activity (inhalation dose)
- External radiation exposure from the activity deposited on the ground (gamma ground dose)

2. Damage appraisal by dose and dose rate computation

The effective dose for the following types of radiation can be computed for **assessing stochastic damage**:

- Gamma submersion
- Ground radiation
- Radiation stress as the result of inhalation on infants (69 years internal radiation exposure)
- Radiation stress as the result of inhalation on adults (50 years internal radiation exposure)

The following organ doses can be determined for **radiation exposure as the result of inhalation** (50 years adults, 69 years children):

- Bone marrow
- Thyroid gland
- Lung (1 year)

The dose rate can also be computed from gamma submersion (from "emitted activity rate").

3. Method of computation

Ground radiation and inhalation are computed on the basis of the incident guidelines.

The following tables (with interpolation) are a fixed part of the program:

- Incident propagation factors for gamma submersion as a function of the distance from the receiving point and the weather category (as specified by the power station).
- Dose factors as a function of the radiation exposure paths, radiation exposure time and nuclide.

The effective **emission levels** have been adopted in the program in accordance with the local conditions as specified by the operator. It is possible to select between chimney and ventilation flaps.

4. Data entry

The following **master data** can be entered via an editor:

- Local coordinates of important receiving points, e.g. village A, X = 5000 m, Y = 8000 m
- Percentage composition of nuclide mixtures
- Absolute composition of nuclide mixtures, each in the following groups
  - Noble gases
  - Iodine
  - Aerosols

The only data which then need to be entered for the individual computations are as follows:

- Weather category
- Wind speed
- Wind direction
- Type of radioactive emission (effective emission level)
- Emitted activity, each in the following groups: Noble gas, iodine, aerosols
- Emitted activity rate, each in the following groups: Noble gas, iodine, aerosols
- Nuclide mixture number, each in the following groups: Noble gas, iodine, aerosols, filtered or unfiltered
- Number of the required receiving point

ENVIRO is intended as an add-on module for the admintec PC program "Radiation-Protection ATLAS" (see poster under this title in this poster session). It computes the results of propagation through the air from constant data (master data) stored in a file and from the current weather and emission data entered either online or offline. The results are displayed on a graphic monitor, by means of colour-coded map sections for instance. Also, after "clicking on a settlement icon with the mouse", it displays the dose or dose rate values computed for this location in a screen window which opens.

5. Hard- and Software Requirements

- IBM Compatible Personal Computer (at least 80386)
- Microsoft Windows 3.1

ESTIMATED MODELS OF IMPACT USED BY *ELECTRICITE DE FRANCE* WITHIN THE FRAMEWORK OF THE FRENCH AND EUROPEAN REGULATIONS

PH. COPPE
Electricité De France
Service Etude Projets Thermiques Et Nucléaires
12-14, avenue Dutrievoz - 69628 Villeurbanne Cedex - France

Summary

Estimated assessments of the impact of a nuclear power station on the environment must be carried out in the regular manner when the installations are in their phase of project, until the stage of their startup.
The estimated models of impact of the gaseous and liquid radioactive releases, used to carry out these evaluations, are the subject of the communication.

1. Objective of the models used ; procedures concerned

During the construction of a nuclear power station, at the time of the phase of project, estimated studies of impact of the releases of radioactive wastes of the installations are necessary. The evaluations carried out appear in the following administrative procedures envisaged by the French and European regulations:

1.1. Request for authorization of creation (DAC)

The request for authorization of creation is deposited for obtaining the Decree of Authorization of Creation which allows the launching of the construction of the nuclear part of the power station.
The file of request for authorization of creation includes, in addition to the description of the projected installations, an impact study and a preliminary safety report describing the provisions taken to ensure the safety and the environmental protection. This file comprises an evaluation of the impact of the radioactive releases in the environment.
The Decree, delivered following the request, fixes on the one hand the regulations to which must conform the plant operator (safety, environnement,...), on the other hand methods of the further procedures (deposit of the Safety Reports and the General Rules of Exploitation) for the startup of the installations.

1.2. Authorization of Releases of radioactive wastes (DAR)

The releases of radioactive effluents in the water and in the air are subjected to specific authorizations which must be delivered before the first tests of the engine of the nuclear power station.
The procedure envisages two stages : preliminary studies deposited at the time of the deposit of the DAC and requests for releases themselves accompanied by the final studies. These files include each an impact study.
The authorizations of liquid releases on the one hand, of gaseous releases on the other hand, are granted by Decrees which fix the limits of the rejected activities, the conditions of releases, the controls imposed to the plant operator and his obligations, in particular with respect to the "Service Central de Protection contre les Rayonnements Ionisants" (SCPRI) of the Ministry in charge of Health.

1.3. Nuclear procedures resulting from the Community legislation (Euratom).

These procedures are written in the accordance with Euratom Treaty (The Treaty of Rome - 1957) which binds the countries of Europe from the point of view of their nuclear installations.

Thus, pursuant to article 37 of the Treaty, EDF prepares a file on the releases of radioactive wastes of the installation concerned. The goal of this procedure is to enable the Commission of the European Communities to determine if the implementation of the corresponding project of release of radioactive wastes is likely to cause contaminations in a Member State other than that where will occur the release. This file, presented to the Commission by the French State, also requires the use of estimated models of impact of the releases of the nuclear power stations.

2. Presentation of the models

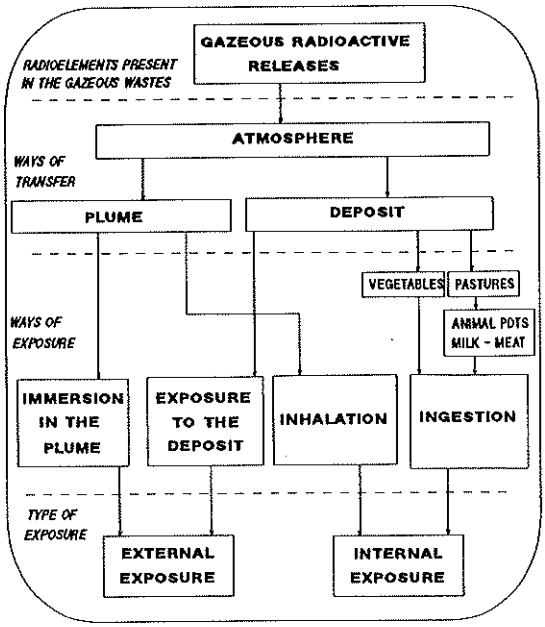
2.1. Types of examined releases ; operating modes.

Releases considered	*DAC-DAR*	*Euratom*
Gaseous Normal operation	x	x
Gaseous Accidental operation		x
Liquid	x	x

The opposite table indicates the types of releases taken into account in the various procedures. The model considered for the accidental releases is specific to procedure Euratom.
The liquid releases taken into account correspond in all the cases to normal operation.

The normal operation of a nuclear power station corresponds to operation to the nominal power with which are associated normal transients of exploitation.
For the accidental release taken into account in file Euratom, one considers the hypothetical accident the most important from the point of view of his radiological consequences : the loss of primary coolant consecutive to a break of the primary cooling system (design basis accident).
Under normal operation, the radioactive bodies likely to be rejected are produced in the core of the reactor: fission products created in fuel and products of activation present in the water of refrigeration.
The liquid wastes are made up of fluids coming from normal leaks related to operation from the installations, movements of primary cooling water associated with the operating transients, washing or draining of circuits likely to be contaminated. The gaseous effluents are mainly produced by the degaseousification of the liquid wastes. The effluents created are treated and stored before release.
At the stage of project, they are wrapping evaluations of the rejected activities, which are considered in calculations of impact in the environment. Thereafter, at the time of the operation of the power station, one checks that the actual values are largely lower. Thus, the liquid effluents taken into account are made up mainly of iodine and of tritium. The radionuclides present in the gaseous effluents are rare gas, iodine, aerosols and tritium.
In the case of the accidental release, studies allow to define the activities which could be released in the atmosphere following the hypothetical accident of loss of primary coolant; the corresponding source term is made up primarily of iodine and of rare gas. They are the values of activities of these radioelements which are retained to carry out the calculation of the levels of corresponding exposure appearing in the file written for the Euratom Treaty.
The paragraphs hereafter present the estimated models of impact used and the factors taken into account. These models were developed by the "Commissariat à l'Energie Atomique" CEA (note in reference [1]).

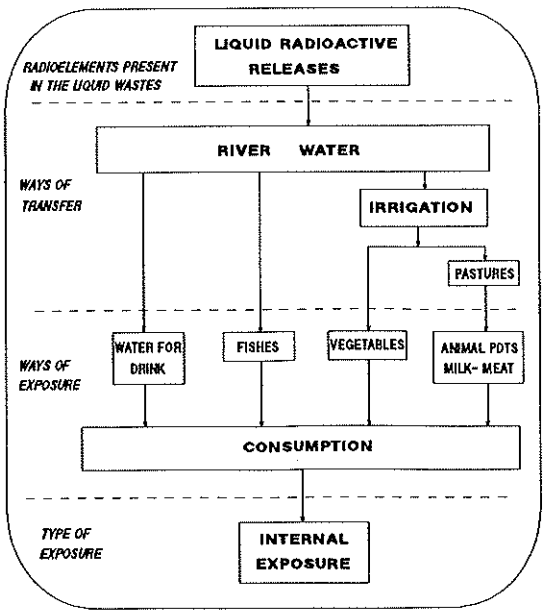
2.2. Impact of the gaseous releases under normal operation.



The figure opposite indicates the mode of taking into account of the transfer of the activity released into the environment : The model calculate for a given place, the concentration of activity in the plume and the activity deposited (ground, plants) ; the levels of external exposure due to the immersion in the plume and the deposit on the ground, as well as the internal level of exposure per inhalation are deduced directly. The exposure of the individual related to consumption takes into account the animal and vegetable products of the feed ration which were subjected to the release. The concentration in radionuclides of the consumed animal products is itself calculated by taking of account the diet of the producing animal considered.

The model evaluates the maximum exposure of the most exposed group of individuals living in the vicinity of the power station (function of the specific weather characteristics of the site).

2.3. Impact of the liquid releases.



The mode of transfer of the activity released into the environment is represented on the figure opposite : the model calculates the concentration of the water of the river and the concentration of the water of irrigation. The activities of water, fish and the irrigated plants are deduced directly. The internal exposure of the individual related to consumption takes into account the animal and vegetable products of the feed ration which were subjected to the release. The concentration in radionuclides of the consumed animal products itself is evaluated by taking into account the diet of the producing animal considered. The model does not calculate the external exposure related to the liquid releases.

Modeling calculates, in the same way that for the gaseous releases, the assessment of the maximum exposure of the group of the most exposed individuals.

2.4. Impact of the gaseous releases under accidental operation.

The model is based on a principle identical to that of the normal releases, however by taking into account the dynamics of the release particular to accidental operation (limited release comprising various episodes). For the immediate consequences, calculation takes account of the ways of external exposure caused by the plume and of internal exposure by inhalation. For the long-term consequences are examined the external exposure to the deposit on the ground and the internal exposure by ingestion of animal or vegetable products.
There are no restrictions on the site of the places of exposure which can be indifferently at short or long distance.

2.5. Models of diffusion.

For the gaseous releases of normal operation, modeling takes into account two classes of diffusion (normal and low) ; the transport and the dispersion of the effluents are in addition evaluated by neglecting the depletion of the plume by deposit on the ground, as well as the radioactive decrease.
In the accidental case, one makes the assumption that the terms of atmospheric dissemination reigning at the beginning of the accident are maintained for all the duration of the release; two coefficients of transfer are considered according to the nature of the radionuclides : rare gas on the one hand and iodine and aerosols on the other hand. The impoverishment of the plume by deposit on the ground is taken into account for the latter. The radioactive decrease in all the cases is considered in the diffusion, taking into account the limited duration of the release.
In the modeling of the liquid releases, one makes the assumption that the releases are distributed in a continuous way in time : the medium flow of the river being known, the annual average concentration is calculated by supposing an immediate mixture of the effluents. The evaluation is identical for the water of irrigation.

2.6. Parameters of the models.

One distinguishes the parameters specific to each nuclear site, and the " standard " parameters valid for all the sites, but which vary according to the radionuclides and (or) introduced food products. The table below summarizes them.

	Gaseous releases	*Liquid releases*
Parameters of site	*- released activities* *- meteorology* *- diets* *- distance to the release point* *- episodes of the release (case of accident)*	*- released activities* *- hydrology* *- diets*
standard Parameters	*- speeds of deposit* *- factors of retention* *- effective average lives* *- ratios of surfaces of the plants* *- factors of translocation* *- factors of Chamberlain (H3)* *- factors of transfer in the animal products*	*- concentration factors* ** in the leafs* ** in the roots* *- decay constants due to the migration* *- concentration factors in fish* *- factors of transfer in the animal products*

3. Developments of the models.

3.1. User-friendliness of the software.

This action in progress in 1992 is related to calculations corresponding to current modelings. It must make it possible to have convivial versions of the computation softwares. It aims at : simplification of the use of the data-processing codes whose data are numerous and complex, homogeneous use of the standard parameters by the users of the codes (values, by default, of certain parameters), improvement of the insurance quality of the codes. This action is, in addition, related to the both others presented in the following paragraphs.

3,2. Evolutions of the models.

A work of development is envisaged on the models described previously : gaseous releases under normal operation for the end of 1992 (work in progress), gaseous releases " accidental " and liquid releases envisaged in 1993. The development is carried out in collaboration with the Commissariat à l'Energie Atomique (CEA/SERE).
Concerning the "normal" gaseous releases, the developed formulation is that of the report in reference [2] ; the principal improvements of this modeling compared to existing calculation concerns the atmospheric model of diffusion, the taking into account of the diets, the calculation of the external exposure due to the deposit on the ground and the treatment of tritium in the transfers. The developed model must allow a more realistic evaluation of the levels of exposure.
For the development of calculations of impact of the accidental gaseous releases, the formulation envisaged is the same one as the preceding one but it takes into account the particular dynamics of this type of release. Except the interest for the procedures that this development presents, the calculation developed will allow a better knowledge of the potential accidental exposures, in particular those related to the hypothetical consequences of the serious accidents of core fusion whose taking into account with the design is carried out in the projects of future reactors.
The choice of the modeling of the impact of the liquid releases to develop is not yet completely fixed currently ; it should probably be carried out starting from the report in reference [3].

3.3. Data base of Parameters.

This action, in progress, is concerned with the development of a file of values (data base) concerning the " standard " parameters evoked previously. The values are referred and relate to a wide field of parameters. The interest of such a development are the coherence of the estimated assessments carried out for the various sites and a better definition of the programs of research and development : starting from an analysis of the vague or missing elements of the data base, the actions (studies, experiments) to undertake to improve it or to supplement it can be carried out. A data base is also planned for the models under development.

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ENVIRONMENTAL IMPACTS OF NUCLEAR INSTALLATIONS CALCULATED USING THE GERMAN FOOD CHAIN MODEL 'AVV ZU §45 STRLSCHV' WITH DECAY CHAINS TAKEN INTO ACCOUNT

Gerald Kirchner
Universität Bremen, FB 1, Postfach 330440, D-2800 Bremen 33

Abstract

Equilibrium-type food chain models usually do not take into account the build-up of radioactive decay products during environmental transport. For estimating the consequences of this simplification, the equations of the terrestrial food chain model of the German *AVV zu §45 StrlSchV* have been extended to include decay chains of variable length. Example calculations show that for releases from light water reactors both during routine operations and in the case of severe accidents the build-up of decay products during environmental transport is of minor importance. However, a considerable number of radionuclides of potential radiological significance are identified which show marked contributions of decay products to calculated radiation dose rates.

1. Introduction

The impact on predictions of food chain models of taking decay chains into account during environmental transport has been analysed recently [1]. In this paper, calculations were based on both model assumptions and parameter values of the terrestrial equilibrium-type food chain model of the *Allgemeine Berechnungsgrundlage* [2]. For licensing procedures in Germany, this model now has been superseded by a revised version [3] showing both some modifications of the model structure and a variety of changes of parameter values. Using the revised model [3], this paper gives an update of the calculations performed previously [1].

2. Calculational Methodology

In contrast to the previous version [2], the model of [3] takes into account downward migration of radionuclides in the unsaturated soil by introducing an effective removal constant λ_k^r from the plant root zone of radionuclide k as

$$\lambda_k^r = \lambda_k + \lambda_k^s$$

where λ_k denotes its physical decay constant and λ_k^s its removal constant from the plant root zone. This change is easily incorporated into the equations of [1] that describe time-dependent activity concentrations in the upper soil layer by replacing the physical decay constant λ_k by the effective removal constant λ_k^r.

The other equations given in [1] for including decay chains of variable length remain unchanged, since there are no further differences in the model structures of the two versions of the terrestrial food chain model. Many of the parameter values, however, have been updated in [3] - in particular those of the transfer coefficients soil → plant, plant → milk and plant → meat, and of human food intake rates.

The calculational procedure described here and in [1] has been incorporated into the ORIGEN-S(UHB) code [4] which is part of a modified version of the SCALE code system [5], as the nuclear-physical data needed for the calculations are easily available within this code system.

3. Example Calculations

Assuming a wet deposition rate of 1 Bq m^{-2} y^{-1} per parent nuclide, Table 1 gives those radionuclides for which decay products built up during environmental transport contribute more than 10 % to the resulting effective dose rate. As it may be noticed, for a variety of radionuclides actual doses to man will be underestimated considerably, if decay chains are not included into the food chain model.

Table 1: Dose rates from individual isotopes resulting from a wet deposition rate of 1 Bq m^{-2} y^{-1} with and without including decay chains during environmental transport [a)]

Isotope	Effective dose rate [μSv y^{-1}] without daughters	with daughters	Fractional contribution of daughter nuclides [b)]
Sr-91	1.96×10^{-5}	4.85×10^{-5}	^{91}Y 59%
-92	3.67×10^{-6}	5.37×10^{-6}	^{92}Y 32%
Zr-93	3.05×10^{-3}	4.21×10^{-3}	^{93m}Nb 28%
-95	3.14×10^{-3}	1.12×10^{-2}	^{95}Nb 72%
Ru-105	1.59×10^{-6}	1.35×10^{-5}	^{105}Rh 88%
Sb-125	3.32×10^{-3}	1.29×10^{-2}	^{125m}Te 74%
-127	3.46×10^{-4}	5.24×10^{-4}	^{127m}Te 28%, ^{127}Te 6%
Te-131m	9.46×10^{-5}	1.04×10^{-3}	^{131}I 91%
-132	2.75×10^{-4}	3.13×10^{-4}	^{132}I 12%
Ba-140	7.48×10^{-4}	1.32×10^{-3}	^{140}La 43%
La-141	1.81×10^{-6}	5.72×10^{-6}	^{141}Ce 68%
Ce-143	4.54×10^{-5}	9.18×10^{-5}	^{143}Pr 51%
Pb-210	$1.11 \times 10^{+1}$	$1.25 \times 10^{+1}$	^{210}Po 12%
Bi-210	2.93×10^{-4}	6.24×10^{-2}	^{210}Po 99.5%
Ra-225	6.44×10^{-2}	8.83×10^{-2}	^{225}Ac 26%
-226	$4.83 \times 10^{+0}$	$1.12 \times 10^{+1}$	^{210}Pb 49%, ^{210}Po 8%
Ac-228	4.40×10^{-6}	3.05×10^{-4}	^{228}Th 44%, ^{224}Ra 49%, ^{212}Pb 5%
Th-227	4.41×10^{-3}	1.37×10^{-1}	^{223}Ra 97%
-228	3.59×10^{-1}	8.02×10^{-1}	^{224}Ra 50%, ^{212}Pb 5%
-229	$3.73 \times 10^{+0}$	$4.43 \times 10^{+0}$	^{225}Ra 12%, ^{225}Ac 4%
-230	5.50×10^{-1}	6.36×10^{-1}	^{226}Ra 5%, ^{210}Pb 7%, ^{210}Po 1%
-232	$2.91 \times 10^{+0}$	$5.98 \times 10^{+0}$	^{228}Ra 47%, ^{228}Th 1%, ^{224}Ra 3%
Pa-231	$1.31 \times 10^{+1}$	$1.54 \times 10^{+1}$	^{227}Ac 14%

[a)] isotopes included if dose rate due to decay products $\geq$10%

[b)] listed if contribution $\geq$1%

This conclusion also holds if dose rates to individual tissues of the human body are calculated as illustrated in Table 2 by the example of ^{232}Th. Additionally, the results of Table 2 emphasize the necessity of including decay products that are built up during environmental transport if interest is in determining doses in areas showing elevated concentrations of members of the natural actinide decay chains (e.g. within the scope of epidemiological studies).

A typical example of effective dose rates calculated for routine atmospheric emissions of a nuclear power plant is given in Table 3, based on emission data of the German

Table 2: Dose rates to human tissues from a wet deposition rate of 1 Bq m^{-2} y^{-1} of ^{232}Th with and without including decay chains during environmental transport

Tissue	Organ dose rate [μSv y^{-1}] without daughters	with daughters	Fractional contribution of daughter nuclides [a]
Urinary bladder	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Breast	5.11×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Upper large intestine	5.89×10^{-2}	$1.35 \times 10^{+0}$	^{228}Ra 82%, ^{228}Ac 1%, ^{228}Th 2%, ^{224}Ra 9%, ^{212}Pb 2%
Lower large intestine	1.69×10^{-1}	$1.79 \times 10^{+0}$	^{228}Ra 66%, ^{228}Th 5%, ^{224}Ra 17%, ^{212}Pb 1%
Small intestine	1.41×10^{-2}	$1.12 \times 10^{+0}$	^{228}Ra 93%, ^{224}Ra 4%
Brain	5.11×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Skin	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Testes	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Bone surfaces	$7.07 \times 10^{+1}$	$1.20 \times 10^{+2}$	^{228}Ra 37%, ^{228}Th 1%, ^{224}Ra 2%
Liver	3.93×10^{-2}	$1.15 \times 10^{+0}$	^{228}Ra 90%, ^{228}Th 1%, ^{224}Ra 3%, ^{212}Pb 2%
Lung	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Stomach	8.25×10^{-3}	$1.10 \times 10^{+0}$	^{228}Ra 95%, ^{224}Ra 3%
Spleen	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Adrenal glands	5.11×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Kidneys	4.71×10^{-3}	$1.09 \times 10^{+0}$	^{228}Ra 95%, ^{224}Ra 3%, ^{212}Pb 1%
Ovaries	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Pancreas	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Red bone marrow	$5.89 \times 10^{+0}$	$1.12 \times 10^{+1}$	^{228}Ra 44%, ^{228}Th 1%, ^{224}Ra 2%
Thyroid gland	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Thymus	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%
Uterus	4.71×10^{-3}	$1.08 \times 10^{+0}$	^{228}Ra 96%, ^{224}Ra 3%

[a] listed if contribution to organ dose $\geq$1%

Wuergassen plant [6] and on the assumptions of (i) 50 % wet deposition, (ii) long-term fallout coefficients of 10^{-9} m^{-2} (aerosols) and 10^{-10} m^{-2} (iodine), and (iii) washout coefficients of 10^{-10} m^{-2} (aerosols) and 5×10^{-10} m^{-2} (elemental iodine). Including decay chains during environmental transport does not affect the total dose rate significantly, as the largest contributions are from nuclides without radioactive decay products.

Effective doses arising from an assumed severe reactor accident are given in Table 4. The radioactive source term which was calculated using the SAS2(UHB)/ORIGEN-S(UHB) code system [4] is based on the assumptions of (i) nuclide inventory of a PWR of the German 1300 MW_e type with core burnup of 22 GWd/t and of (ii) core release fractions equivalent to the PWR1 release category of the U.S. Reactor Safety Study [7]. Deposition is assumed to occur by rain 1 d after release with washout coefficients of 10^{-7} m^{-2} (aerosols) and 5×10^{-7} m^{-2} (elemental iodine). It should be remembered that radionuclide transport after a single deposition event can be modeled using an equilibrium-type food chain model [1]. Again, doses arising from decay daughters built up during environmental transport are of importance for a variety of isotopes, but give only minor contributions to the total dose.

Table 3: Calculated dose rates from routine emissions of the Wuergassen nuclear power plant taking decay chains into account during environmental transport

Isotope	Emission rate [a] [Bq y^{-1}]	Effective dose rate [nSv y^{-1}]	Contribution of daughters [%]	Isotope	Emission rate [a] [Bq y^{-1}]	Effective dose rate [nSv y^{-1}]	Contribution of daughters [%]
Mn-54	1.7×10^{6}	1.8×10^{-3}	0	Cs-134	1.2×10^{7}	9.7×10^{-1}	0
Co-58	9.0×10^{5}	9.9×10^{-4}	0	-137	6.4×10^{7}	$4.9 \times 10^{+0}$	0
-60	7.6×10^{7}	$1.2 \times 10^{+0}$	0	Ba-140	2.9×10^{7}	1.7×10^{-2}	43.
Zn-65	4.5×10^{6}	1.5×10^{-1}	0	La-140	1.8×10^{7}	8.1×10^{-4}	0
Sr-89	4.6×10^{7}	9.2×10^{-2}	0	Ce-141	4.8×10^{6}	1.6×10^{-3}	0
-90	2.3×10^{6}	4.0×10^{-1}	6.5	-144	2.0×10^{6}	1.6×10^{-2}	0.6
Zr-95	1.0×10^{5}	4.9×10^{-4}	72.	Pu-238 [b]	1.5×10^{4}	2.0×10^{-2}	0
Nb-95	3.2×10^{6}	1.5×10^{-2}	0	-239 [c]	5.0×10^{3}	7.2×10^{-3}	0
Ru-103	2.6×10^{6}	1.1×10^{-3}	0.8	Cm-242	4.9×10^{4}	1.8×10^{-3}	4.5
-106	1.5×10^{6}	1.6×10^{-2}	0.1	-244	2.0×10^{4}	1.6×10^{-2}	0
I-131	8.0×10^{8}	$9.0 \times 10^{+0}$	0	Total		$1.7 \times 10^{+1}$	0.2

[a] taken from ref. [6]
[b] $^{238}Pu + {}^{241}Am$
[c] $^{239}Pu + {}^{240}Pu$

4. Conclusions

As the example calculations indicate, neglecting decay daughters that are built up during environmental transport in the terrestrial food chain model of the *AVV zu §45 Strahlenschutzverordnung* underestimates doses from nuclear reactors only slightly both during routine operations and in the case of severe accidents. On the other hand, our calculations identify a variety of radionuclides of potential radiological significance for which decay chains during environmental transport should be taken into account. Comparison with the conclusions [1] based on a previous version [2] of the food chain model shows that the numerous modifications of the revised version [3] do not alter results significantly.

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Table 4: Calculated effective doses from a hypothetical core melting accident of a nuclear power reactor [a]

Isotope	Activity released [Bq]	Effective dose [Sv]	Contribution of daughters [%]	Isotope	Activity released [Bq]	Effective dose [Sv]	Contribution of daughters [%]
Co-58	3.7×10^{16}	$9.3 \times 10^{+0}$	0	I-132	3.7×10^{18}	4.2×10^{-1}	0
-60	1.4×10^{16}	$5.2 \times 10^{+1}$	0	-133	5.5×10^{18}	$1.0 \times 10^{+2}$	0
Rb-86	7.0×10^{14}	2.4×10^{-1}	0	-135	5.1×10^{18}	$1.2 \times 10^{+0}$	0
Sr-89	2.0×10^{17}	$9.2 \times 10^{+1}$	0	Cs-134	1.3×10^{17}	$2.6 \times 10^{+3}$	0
-90	9.7×10^{15}	$6.6 \times 10^{+2}$	6.5	-136	8.8×10^{16}	$3.1 \times 10^{+1}$	0
-91	2.4×10^{17}	2.0×10^{-1}	60.	-137	1.1×10^{17}	$2.4 \times 10^{+3}$	0
Y-91	1.5×10^{16}	$6.9 \times 10^{+0}$	0	Ba-140	3.4×10^{17}	$4.3 \times 10^{+1}$	43.
Zr-95	2.0×10^{16}	$2.2 \times 10^{+1}$	72.	La-140	2.1×10^{16}	$1.3 \times 10^{+0}$	0
Nb-95	1.9×10^{16}	$2.2 \times 10^{+1}$	0	Ce-141	2.0×10^{16}	$1.5 \times 10^{+0}$	0
Mo-99	2.9×10^{18}	$4.4 \times 10^{+1}$	0.6	-143	1.8×10^{16}	1.0×10^{-1}	51.
Ru-103	2.4×10^{18}	$2.3 \times 10^{+2}$	0.8	-144	1.2×10^{16}	$2.2 \times 10^{+1}$	0.6
-106	5.7×10^{17}	$1.4 \times 10^{+3}$	0.1	Pr-143	1.8×10^{16}	8.2×10^{-1}	0
Rh-105	9.8×10^{17}	$7.2 \times 10^{+0}$	0	Nd-147	8.1×10^{15}	2.5×10^{-1}	3.8
Sb-124	3.0×10^{14}	1.8×10^{-1}	0	Np-239	2.4×10^{17}	9.7×10^{-1}	1.7
-125	7.1×10^{15}	$1.1 \times 10^{+1}$	74.	Pu-238	1.4×10^{13}	$4.4 \times 10^{+0}$	0
-127	1.0×10^{17}	$4.5 \times 10^{+0}$	34.	-239	3.8×10^{12}	$1.3 \times 10^{+0}$	0
Te-125m	1.4×10^{15}	$1.5 \times 10^{+0}$	0	-240	3.8×10^{12}	$1.3 \times 10^{+0}$	0
-127m	1.2×10^{16}	$4.4 \times 10^{+1}$	7.8	-241	1.1×10^{15}	$6.9 \times 10^{+0}$	2.6
-129m	5.3×10^{16}	$8.5 \times 10^{+1}$	1.2	Am-241	8.0×10^{11}	2.8×10^{-1}	0
-131m	1.7×10^{17}	$1.0 \times 10^{+1}$	99.	Cm-242	2.4×10^{14}	$2.0 \times 10^{+0}$	4.5
-132	1.6×10^{18}	$4.2 \times 10^{+1}$	12.	-244	9.5×10^{12}	$1.8 \times 10^{+0}$	0
I-131	2.6×10^{18}	$3.6 \times 10^{+3}$	0	Total		$1.2 \times 10^{+4}$	1.0

[a] isotopes included if dose $\geq 10^{-1}$ Sv

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RADIOLOGICAL INFORMATION SYSTEM FOR REGIONAL MONITORING AUTHORITIES *"KERNREAKTOR-FERNÜBERWACHUNG"* SYSTEM - DISPERSION AND DOSE CACULATIONS -

Brücher,L., Siemens AG, Power Generation Group (KWU),Offenbach/M., Germany
Salfeld,H.Ch., Niedersächsisches Landesamt für Immissionsschutz, Hanover, Germany

Summary: The paper describes several specific functions of the radiological information system for nuclear power plant environmental monitoring, such as automatic alarm system, mobile monitoring of dose rates and Lagrangian dispersion calculations.

1 Introduction

Operators of nuclear power plants are required by law to report accidents and other safety-related incidents to government monitoring authorities. This includes in particular any discharge of radioactive substances to the environs exceeding the limits allowed for normal operation and anticipated operational occurrences. In Germany, state monitoring authorities are equipped with a system for regional environmental monitoring called "Kernreaktor-Fernüberwachung" (KFÜ) which provides plant-specific data - particularly those regarding activity releases - directly to the monitoring authorities.
In the German state of Lower Saxony, measured data from the region's four nuclear power plants are collected on site at substations and transmitted via the DATEX public telecommunications network to the state monitoring authorities' main data processing station in Hanover. The measured data are updated at 10-minute intervals and stored for further processing. The TIS information system (acronym based on the German name for "Dispersion and Environmental Monitoring System for Calculating Doses") used within the KFÜ primarily serves to calculate potential radiation exposures in plant environs and to analyze the measured data. For this purpose, the system provides cyclic acquisition of data on radioactive discharges, atmospheric dispersion conditions and dose rates resulting from direct exposure measured at several locations in the plant environs.
The TIS information system essentially serves to perform the following tasks in support of accident management efforts, for emergency preparedness exercises and for the regular monitoring activities carried out by the state authorities:

- Redundant storage and controls of measured data (radiological, meteorological and environmental data)
- Automatic alarm system to inform accident management personnel in the event that specified limits are exceeded.
- Calculation of potential radiation exposure of persons in the plant environs for all exposure pathways taking into consideration data from nuclide-specific measurements. TIS offers two dispersion models:
  * Gaussian dispersion model in accordance with the standards (Ref. 5)
  * Time-dependent Lagrange particle dispersion model (Ref. 4)
- Statistical analyses of measured data, e.g. dispersion statistics, time histories, bar charts, etc.

TIS is used at a number of nuclear power plants as well as for state-wide monitoring activities in Lower Saxony, and beginning in 1993 a similar system will be operational in Switzerland. In this report, several specific functions performed by KFÜ in Lower Saxony are introduced which demonstrate

how the emergency preparedness of KFÜ is ensured and how its area of application can be considerably broadened through the integration of environmental monitoring and advanced dispersions models.

2 Alarm Annunciation

The KFÜ regional monitoring system is equipped with an automatic alarm system by which alarms are annunciated by telephone and monitoring personnel informed of the given situation. A distinction is made between the following two objectives:

- Radiological Alarm
  KFÜ monitors radiological discharges from nuclear power plants using redundant systems of measuring equipment (operator- and state authority-dedicated units) to ensure compliance with specified limits such as the approved daily emissions limits.
- Technical Alarm
  The technical alarm system serves to check for fault-free functioning of the measuring equipment and data transmission from the substations to the nuclear power plants as well as to monitor the programs operating on the KFÜ main station computer which are necessary to ensure proper monitoring.

Based on a freely-selectable logic connection of events and conditions, identifiable through binary signals (e.g. failure of monitor xy) or measured and computed values, binary alarm signals (time-delayed as required) can be transmitted to a TELENOT type automatic alarm module. This unit initiates an alarm at the main station and also informs emergency personnel in the field via telephone or the Euroruf paging system.

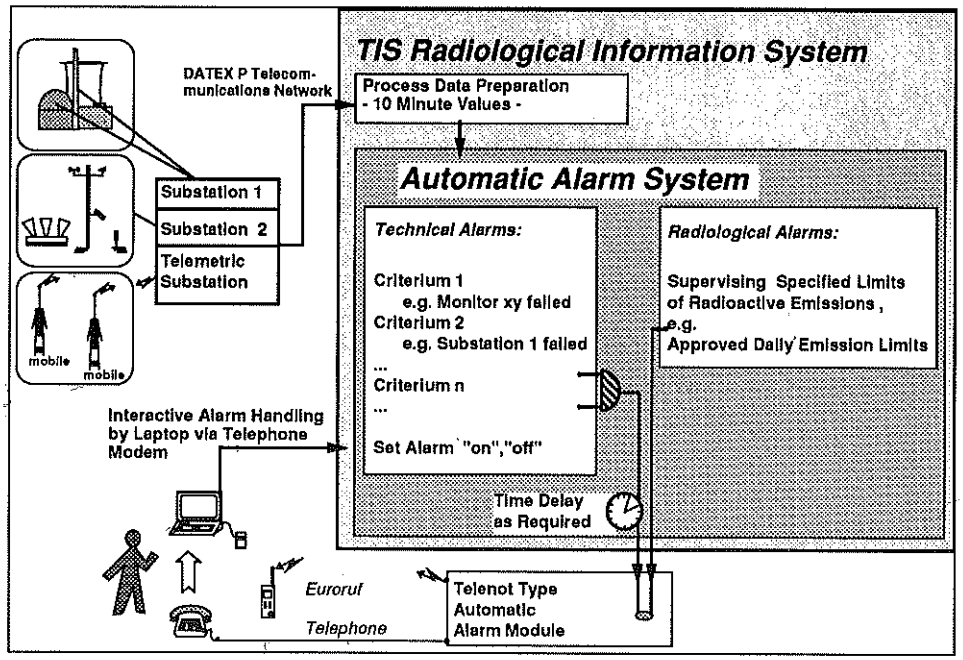

Fig. 1: The radiological information system provides an automatic alarm system

Personnel on duty at the main station can access the KFÜ's information system, execute a rapid analysis of the alarm to determine its nature (e.g. failure of monitor xy) using special time histories and records, and immediately implement appropriate countermeasures. Incoming technical alarms can be acknowledged and canceled by the monitoring personnel.

Personnel notified in the field can dial into the information system at the main station using a laptop computer equipped with a modem. Data protection is ensured by the target modem at the main station computer. Once notified, the target modem attempts to reestablish the connection to the field unit according to a specific code, switchover allowing access to the main computer not being effected until the connection (and thus caller identification) has been successfully established.

3 Mobile and Stationary Monitors of Environmental Gamma Dose Rate

The actual gamma dose rate resulting from direct exposure in the environs of a nuclear power plant is determined by a network of up to 30 mobile dose rate monitors (Ref. 7). The measured data acquired by these battery-driven monitors are transmitted by radio to substations located at the nuclear power stations from where the signals are supplied via the DATEX network to the main station. In the event of an accident the monitors are set up in the plant environs at previously defined locations. They can be relocated in the course of a monitoring campaign to allow optimum positioning of the monitors to be achieved, for example, in the event of a change in the weather conditions over a large area entailing a change in the prevailing wind direction.

The definition of possible locations is effected via a user interface menu, although user input errors (e.g. multiple application of a single monitor) are prevented by a dedicated logic system.
The actual locations of the monitors during a monitoring campaign are recorded by the user via a user interface in the information system. The dose rate monitors are not provided with any capability for automatic identification of location, but such a function can be backfitted without difficulty thanks to the data model which is very variable.

Continuous dose rate monitoring and checks of the radio network are performed by several mobile monitors which remain permanently in the field, while the remaining monitors are centrally stored until required.

To supplement this mobile system, measured data from the stationary monitors belonging to the monitoring network (WADIS system) operated by the Bundesamt für Zivilschutz, Germany's federal bureau of civil defense, are also fed into Lower Saxony's KFÜ regional monitoring system and made available for analysis (Ref. 3).

4 Dispersion Calculations Using A Lagrangian Particle Model

The most important task performed by the information system is the calculation of dose distributions resulting from the dispersion of airborne emissions as a function of space and time for potential effective doses and partial body doses (Refs. 1 - 3). The dose model defined in the regulatory guidelines named below as Reference 5 serves as a basis. TIS offers two fundamentally different dispersion models for simulating the transport and

dispersion of airborne materials. A quasi-time-dependent Gaussian dispersion model such as the one described in Reference 5 is standardly used in which the doses of the individual time periods are combined to obtain the total dose.
In addition, the Lagrange particle dispersion model (LASAT) is used (Ref. 4). This model simulates the dispersions of radioactive substances in the atmosphere as a function of space and time. Nuclide-specific distribution in the exhaust gas path is simulated using groups of representative particles. The transport of particles is calculated by means of a deterministic displacement of the particles corresponding to the average wind patterns, while dispersion is determined on the basis of a stochastic displacement (Markov process) of the particles corresponding to the given atmospheric turbulence. Based on particle density, LASAT calculates variations in substance concentration, fallout and washout as well as gamma submersion. This allows nuclide-specific measurements of radioactive discharges to be taken into consideration. The dose distributions for all exposure pathways can be calculated from this data using the TIS dose model.

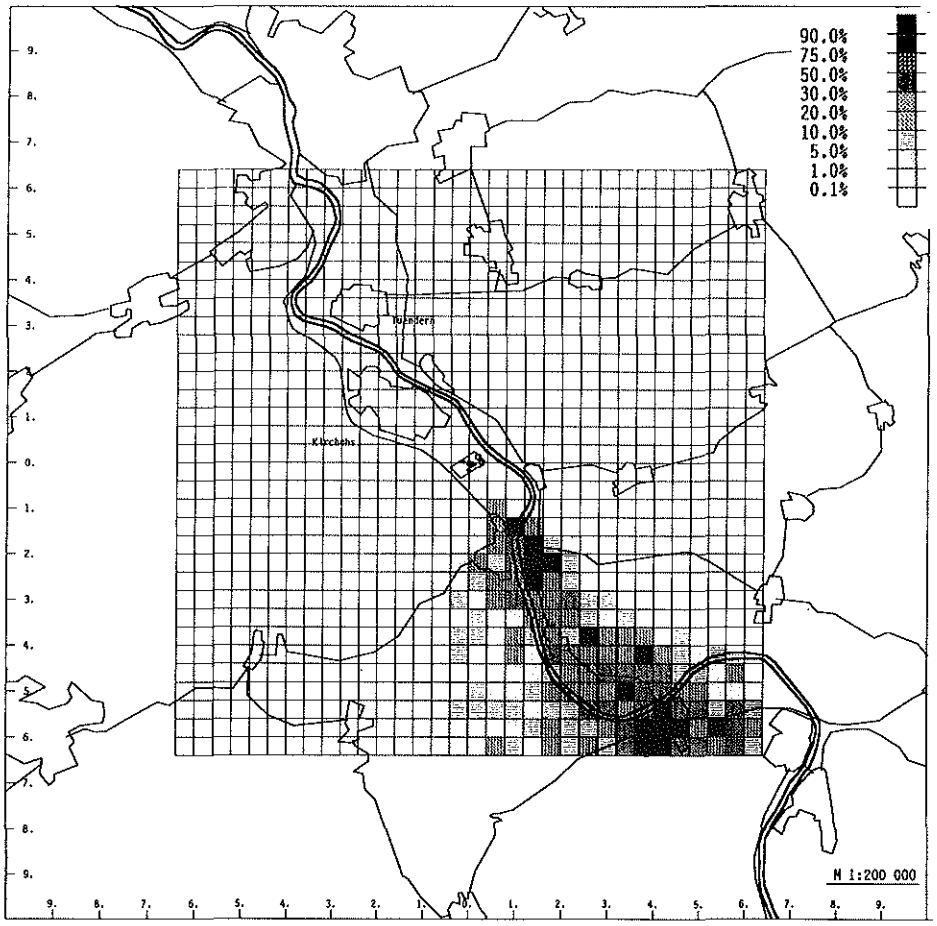

Fig. 2: Dose distribution as a function of space and time, resulting from the dispersion of airborne emissions, calculated with the Lagrangian dispersion model and the TIS-dose model.

In order to receive results in a 10 minute cycle, the LASAT program runs on a transputer which communicates via an Ethernet connection with the Unix computers at the main station. TIS supplies the LASAT dispersion calculation system with measured data, constants and control variables and stores the LASAT results. TIS can thus be used to perform various analyses for the modeled time period without renewed LASAT computation.
Upon fulfillment of specific criteria (radioactive discharges exceeding emissions limits) the TIS-LASAT system switches over to a real-time operating mode. From the incoming measured data, provided in 10-minute resolutions, continuously updated dispersion and dose distributions are calculated. These analysis results are stored in a corresponding memory which can be continuously accessed by the user.

5 Conclusion and Outlook

The emergency preparedness of the KFÜ system is an essential prerequisite to proper execution of the monitoring task; preparedness is efficiently supported by the automatic alarm annunciation system.The incorporation of mobile and stationary local dose rate monitors allows accurate, accident-specific assessment of the given situation. Through this system, in conjunction with the particle dispersion model as a basis for calculating potential radiation exposures, a solid basis is achieved for the use of KFÜ within the framework of accident management.

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THE OPERATIONAL E.D.F PUFF MODEL : "DIFBOU"

D.WENDUM

Electricite de France. Direction des Etudes et Recherches. Département Environnement.
6, quai Watier 78401 Chatou Cédex. FRANCE

ABSTRACT

We present the equations and ways of computing transport-diffusion of radioactive pollutants in the atmosphere by a gaussian puff technique implemented in the DIFBOU program developed by the group Meteorology and Climat of Electricité de France for operational purposes on nuclear sites: the computation in real time of the radiological consequences of an accidental release of radioactive pollutants in the atmosphere. The installation of this program on different nuclear sites of EDF has begun.

1.GENERAL PRESENTATION

The goal consists in being able to compute in real time the consequences of an accidental or normal release of radioactive pollutants in the atmosphere by a nuclear power plant. By consequences is meant physical values: air concentrations, cumulated depositions and estimates of the doses received by the population.
The transport and diffusion in the atmosphere of the pollutants is modelled by a superposition of puffs. The center of mass of the puffs moves with the wind and the size of the puffs grows with time.
At a given time the available meteorological data consists in one vertical profile of the horizontal wind, the stability of the atmosphere and the precipitation rate.
It is assumed that on the computational domain (10 to 30 km around the power plant) the same meteorological situation prevails. Hence it is meaningless to use this model for the estimation of the consequences of a release at large scale (50 km or more).

2. THE EQUATIONS OF THE MODEL

2.1. Transport

The wind is taken into account by moving the center of mass of the puffs withs speed equal to the wind speed. In the horizontal direction the wind can be split in w-e and s-n componants u(z,t) and v(z,t). The trajectory of the center of mass has then equations:

$$\frac{dx}{dt} = u(z,t) \qquad \frac{dy}{dt} = v(z,t)$$

Since no data concerning the vertical wind speed is available the center of mass remains at constant relative altitude z above the topography.

2.2. Diffusion

An instantaneous and punctual release of a mass M of pollutant will progressively be diluted in the atmosphere. If the release takes place at (0,0,0) and time t=0, the quantity of pollutant at (x,y,z) and time t>0 will be:

$$C(x,y,z,t) = \frac{M}{(2\Pi_2^{\frac{3}{2}}\sigma_H^2\sigma_V)} \exp(-(x^2+y^2)/(2\sigma_H^2) - z^2/(2\sigma_V^2))$$

where $\sigma_h=\sigma_h(t)$ and $\sigma_v=\sigma_v(t)$.

This is the expression of a gaussian puff, with stantard deviations σ_h (horizontal) and σ_v (vertical). They are empirical functions of the travel time of the puff, and depend on the stability of the atmosphere. The C.E.A (French Atomic Agency) /1/ recommends the use of DOURY's laws based on a classification in two categories of the stability of the atmosphere (weak diffusion or normal diffusion).

2.3. Superposition

In the precedent paragraph we have dealt with an instantaneous and punctual release, and did not take the wind into account. In the case of a continuous release one must add the different contributions of the different puffs.Letting (0,0,h) being the point of release, D(t) the release rate (dimension= quantity of pollutant per unit time), then assuming the release started at t=0 the concentration of pollutant at (x,y,0) (ground level) has at $t > 0$ the following expression:

$$C(x,y,0,t)=\int_0^t \frac{D(\theta)}{(2\Pi^{\frac{3}{2}}\sigma_H(t-\theta)^2\sigma_V(t-\theta))} \exp(-((x-x_c)^2+(y-y_c)^2)/(2\sigma_H(t-\theta)^2)-h^2/(2\sigma_V(t-\theta)^2))d\theta$$

The integration with respect to θ means that we sum up all the contributions to the concentration at (x,y,0) of all the puffs released between 0 and t. $t-\theta$ is the travel time of the puff released at time θ and carrying the mass $D(\theta)$.
The center of mass (X_c, Y_c) of the puff has the expressions:

$$x_c=x_c(t,\theta)=\int_0^t u(h,\tau)d\tau \qquad y_c=y_c(t,\theta)=\int_0^t v(h,\tau)d\tau$$

2.4. Radioactive decay

In order to take it into account it suffices to replace the factor $D(\theta)$ by $D(\theta)2^{\frac{(t-\theta)}{T}}$ where T is the half-life of the radioactive material in the integral defining C(x,y,0,t).

2.5 Ground reflection

The tail of the puffs wich is below the ground level (z<0) is reflected above the ground level, leading at z=0 simply to a doubling of the concentrations. It will therefore suffice to multiply the integral by 2.

2.6 Reflection at the top of the mixing layer

At the scale the model is supposed to deal with (10 to 20 km around the power plant) we may neglect this phenomena. The influence of the mixing layer becomes noticeable only at larger scales.

2.7 Deposition

Our model must not only take into account transport, diffusion, radioactive decay and ground reflection but also dry and wet deposition on the soil.

This taking into account splits in two independant problems:

- diminishing the mass of the puffs (depletion)

- spreading on the ground the mass lost by the puffs

2.7.1 Depletion

2.7.1.a / Wash out by precipitations

L(t) being the precipitation rate we consider that the mass lost by the puff between t and t+dt has the following expression :

$dM(t)=c_l L(t)M(t)dt$ (M(t) is the mass of the puff, c_l a wash-out coefficient).

2.7.1.b / Dry deposition

Dry deposition is taken into account with the use of a coefficient v_c(deposition velocity).We consider that the mass lost by the puff between t and t+dt is now :

$$dM(t)=\frac{2v_d M(t)}{\sigma_v\sqrt{2\Pi}}exp(\frac{h^2}{2\sigma_v{}^2})dt$$

Adding all the removed masses the total effect of radioactive decay, dry and wet deposition is to replace the factor $D(\theta)$ in the integral defining $C(x,y,0,t)$ by $F(t,\theta)$ verifying the following equations:

$$F(\theta,\theta)=D(\theta)$$

$$\frac{\partial F}{\partial t}(t,\theta) = -(\frac{Log(2)}{T} + c_l L(t) + \frac{2v_d}{\sigma_v(t,\theta)\sqrt{2\Pi}} exp(-\frac{h^2}{2\sigma_v(t,\theta)^2})) F(t,\theta)$$

Here t is the time at wich C(x,y,0,t) is computed, and θ the time of release of pollutant.

2.7.2 Ground repartition of deposited pollutant

At time t the total mass deposited by the pollutant released at time $\theta < t$ is :

$$d(t,\theta) = F(t,\theta)\ (c_l L(t) + \frac{2v_d}{\sigma_v(t,\theta)\sqrt{2\Pi}} exp(-\frac{h^2}{2\sigma_v(t,\theta)^2}))$$

We spread this mass on the ground with a 2D gaussian density having center of mass (Xc,Yc) and standard deviations σ_h (t, θ).
The quantity of pollutant deposited per unit surface at time t, location (x,y) by the pollutant released at time θ has then the following expression:

$$dep(x,y,t,\theta) = \frac{d(t,\theta)}{2\Pi\sigma_h(t,\theta)^2} exp(-\frac{(x-x_c(t,\theta))^2+(y-y_c(t,\theta))^2}{2\sigma_h(t,\theta)^2})$$

The total amount deposited at time t, location (x,y) is then obtained by integrating all the contributions of all the emissions between 0 and t, hence:

$$dep(x,y,t) = \int_0^t \frac{d(t,\theta)}{2\Pi\sigma_h(t,\theta)^2} exp(-\frac{(x-x_c(t,\theta))^2+(y-y_c(t,\theta))^2}{2\sigma_h(t,\theta)^2})d\theta$$

The **total amount of pollutant** (dimension= "quantity of pollutant" per unit surface) **found on the ground** at time t is then the sum of all the quantities deposited between 0 and t. With due respect paid to radioactive decay it has then the following expression:

$$Q_{sur}(x,y,t) = \int_0^t 2^{-\frac{t-\tau}{T}} \int_0^\tau \frac{d(t,\theta)}{2\Pi\sigma_h(t,\theta)^2} \exp\left(-\frac{(x-x_c(t,\theta))^2+(y-y_c(t,\theta))^2}{2\sigma_h(t,\theta)^2}\right) d\theta \, d\tau$$

2.8 Growth of the puffs

The rules wich will be presented are those recommended by the Frenc Atomic Agency (C.E.A) /1/, based on a classification of the stability of the atmosphere in two classes: the DOURY classes.The laws of growth of the standard deviations σ_h (horizontal) and σ_v (vertical) are quite different :

- σ_h is an empirical function of the age of the puff, given A PRIORI. This function does not depend on any meteorological parameter.
- σ_v has a law of growth depending on the stability of the atmosphere.For both Doury classes of stability (normal or weak diffusion) an empirical curve giving the standard deviation as function of the age of the puff have been established. So if stability does not change with time we have $\sigma_v = \sigma_v(\, t, \text{"normal"})$ or $\sigma_v = \sigma_v(\, t, \text{"weak"})$,t being the age of the puff. In the case that the stability changes with time the calculation goes as follows.

Suppose that before t_c we have normal diffusion and after t_c weak diffusion so that at time t_c we have $\sigma_v(t_c) = \sigma_v(\, t_c, \text{"normal"})$.

Let t_c^0 be that (fictitious) age giving the same standard deviation in weak diffusion conditions so that we have $\sigma_v(\, t_c^0, \text{"weak"}) = \sigma_v(\, t_c, \text{"normal"})$. As long as the stability does not change for $t > t_c$ the standard deviation will be expressed as $\sigma_v(\, t\,) = \sigma_v(\, t_c^0 + (t-t_c), \text{"weak"})$. The same computation will be done if at some other time t_d the stability goes back to normal diffusion conditions. That is defining t_d^0 so that $\sigma_v(\, t_d\,) = \sigma_v(\, t_d^0\,, \text{"normal"})$ for $t > t_d$ the standard deviation will be expressed as $\sigma_v(\, t\,) = \sigma_v(\, t_d^0 + (t-t_d), \text{"normal"})$.

2.9 Effective activity of a species

According to what we already presented our model does not take into account the descendants of a radioactive species (we do only take into account radioactive **decay**). For doing that the French Atomic Agency (C.E.A) recommends to replace the standard decay equation

$$\frac{dA}{dt} = -\frac{Log2}{T} A \quad \text{(T is the half life of the species)}$$

by the following equation

$$\frac{dA}{dt}(t) = -\frac{Log2}{T(t-t_{stop})} A(t)$$

where T(.) is a precomputed function of the time elapsed since the stopping of the nuclear reaction in the reactor. This function T(.) will then be characteristic of a family of radioactive

elements (initiators + descendants).So now it will suffice to replace in all the equations already given :

- the terms $\frac{Log2}{T}$ by $\frac{Log2}{T(t-t_{stop})}$

- as well as $2^{-\frac{(t-\tau)}{T}}$ by $\exp\left(-\int_{\tau}^{t} \frac{Log2}{T(\theta-t_{stop})} d\theta\right)$

2.10 Dose calculation

The dose exposure rate depends on:

- the activity present in the air (irradiation by the cloud)
- the activity on the ground (irradiation by deposited material)

The French Atomic Agency /1/ recommends the following formulation for the dose exposure rate at point (x,y) at ground level and time t :

$$d.e.r\ (x, y, t)=\alpha(t-t_{stop})A_{air}(x, y, t)+\beta(t-t_{stop})A_{ground}(x, y, t)$$

where α and β are empirical functions of the time elapsed since the stop of the nuclear reactor in the power plant. They are tabulated for each species of radioactive material and each relevant biological tissue (thyroid, lung, etc...). In order to calculate the dose received at a given location one has then to integrate with respect to time the dose exposure rate.

2.11 Numerical calculations

The principles exposed do not lead to very complicated calculations. We first face the problem of integrating differential equations whose solutions give us the position, standard deviation, activity and deposited activityof each puff. Then we must calculate at each relevant point (generally points on a cartesian grid enclosing the power plant) the contribution to the activity at that point of each puff present on the computational domain. If for each grid point we sum the contribution of ALL the puffs the CPU time required increases beyond any reasonable limit (20 min CPU on IBM 3090 for simulating 12 hours of release on 31x31 grid).To avoid this excessive consumption of computing time the domain of influence of each puff has been restricted : we compute the contribution of a puff only at those points situated at a distance =< 4 σ of it's center of mass.By doing this the CPU time of the previous test case fell to 10-20 seconds.

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ATMOSPHERIC DISPERSION OF RADIOACTIVITY IN COMPLEX TERRAIN: MODEL EVALUATION AND SELECTION FOR REAL TIME EMERGENCY APPLICATIONS

V. R. D. Herrnberger, P. Doria, ASH/PSI, CH-5232 Villigen PSI
G. Prohaska, HSK, CH-5232 Villigen-HSK

Summary

A four phase model evaluation scheme is described to evaluate and select models out of the large number of available software packages, which are well suited for real time dispersion simulation in emergency situations. Starting from a brief review of the state of the art of atmospheric dispersion modelling of radioactive/inert substances in complex terrain the way to select best referenced models is described. They are practically tested by applying them to the analysis of the tracer experiment SIESTA over the Swiss plane. The results are discussed and general conclusions are drawn with respect to the dispersion model comparison and selection.

Introduction

The influence of complex terrain on atmospheric dispersion processes of airborne material is important in the local and lower meso scale range. Its modelling in the real time or prognostic mode is still a challenge for modern scientific computing, because of the complexity of the involved phenomena, their different space and time scales and the large amount of topographical and meteorological data. In this paper the research subject is limited to the modelling of the dispersion process only, assuming, that all necessary wind and turbulence data will be available from ongoing research (e.g. in the region of Bale) or from meteorological monitoring networks.
Modern know-how for solving such physical problems of more general interest is accumulated worldwide in numerous models and codes. The compatibility of modern programming languages in connection with fast computer networks facilates the transportability, implementation and efficient use of software and data bases over large distances. Therefore a project manager can make up his mind for a cost/benefit analysis to find the better solution of his problem
- either by developing model and software on the basis of open literature
- or by screening the available models and software packages.
Evidently both ways have to be persued in respect of the requirements of the project. In our case of atmospheric dispersion modelling in complex terrain we followed the second way. On one hand much development is already done or underway in the air pollution field. On the other hand we estimated the cost/benefit ratio to 1:10, because 10 man years development for such a code is not exceptional. However model evalution method and performance measure (What is the "best" model/code for the problem?) is one of the most difficult task. Different techniques of model performance measure are developed and employed. None is generally approved or accepted up to now (Munn, 1987).

1) Review phase

Here the state of the art, type of models and their approximations are reviewed by a questionnaire to the modellers. It covered about 90 items to characterize the models. Four different approaches were found:

1) In the *Langrangian particle* method the advection and dispersion is simulated by random walk (Monte Carlo). The results contain inherent statistical errors.
2) In the *Eulerian grid* method the process is approximated by an advection/diffusion differential equation, which is integrated numerically by discretization in space and time (K-theory). The results contain inherent errors from "numerical diffusion".
3) In the *Gaussian puff* approach the cloud is simulated by a time series of puffs. Their growth and inner concentration pattern follow a Gaussian distribution.
4) Modern *fluid dynamic processors* contain either Lagrangian or Eulerian type dispersion models and are recently applied to atmospheric dispersion problems. In general they are advanced in turbulence simulation and numerical iteration schemes.

2) Pre-evaluation and -selection phase

Here the questionnaire completed by the modellers are evaluated and screened for few "best referenced" models, to undergo the practical test of phase 3.
A semi-quantitative evaluation with respect to
- scientific performance
- soft- and hardware performance
- user-friendliness

was done by attributing merit points for the quality of the modeller's answer per item, Table 1.

Tab. 1: Example for the pre-evaluation and -selection phase

Dispersion Code Capabilities & Evaluation VI			LAGR_N	p	LAGR_N		EULER_2	p	PUFF_N	p	PUFF_2	p	NHEL_N	p	w
- Implementation	1.6.22	Programming language	FORTRAN	2	F77	2	F77	2	F77	2	F77	2	FORTRAN	2	2
	1.6.23	PC compatible	Y	2	(Y)	1	N	0	(Y)	1	(Y)	1	N	0	1
	1.6.24	on computer/under OS	V,H !	2	C,V,I,H/U	2	C,V,I/U	2	C,V,I/U	2	C,V,I/U	2	C,V,S,I	2	1
	1.6.25	Effort for VAX/CONVEX (MW)	N	2	N	2	small	2	small	2	small	2	N	2	2
	1.6.26	Graphic software	CA-DIVX11	2	DISSPLA	2	DISSPLA	2	DISSPLA	2	DISSPLA	2	GKS	2	2
7. Variat	1.7.1	Related codes		0	MESSNEW	2	ADSO/MESS	2		0	HERM/MINE	2	N	0	1
	1.7.2	Related to		0		0		0		0		0	N	0	0
	1.7.3	- R & D	Y	2	Y	2	Y	2		0	Y	2	Y	2	2
	1.7.4	- Industry/NPP		0	Y	2	-	0	Y	2	Y	2	Y	2	1
	1.7.5	- Safety board		0	Y	2	Y	2	Y	2	Y	2	Y	2	2
Sum of meritpoints, 'Σ p'			96/80		93/74		107/90		105/85		114/92		101/87		
Weighted sum of meritpoints, 'Σ (w x p)'			245/202/19		204/166/41		262/220/26		253/199/37		262/212/41		213/182/37		

Legend	Y = yes, (Y) = conditional, n = no, empty = to be agreed or unknown, or give special information
- Transport/Flow model:	E = Eulerian, L = Lagrange, P = Gaussian puff / IN = interpolating, D = Diagnostic, H = Hydrostatic, NH = Non-hydrostatic
- Hardware:	C = Cray, H = HP, I = IBM, S = Sun, U = UNIX/UNICOS, V = Vax, TR = Transputer
- Comments:	(OP) = conditional operational, Ø = no dose model, ! = answer from interview/documents, ? = open points, Do = doubts
- Meritpoints/Weights 'p'/'w',	2 = very good, 1 = limited performance, blank = not considered / 4 = very relevant, 2 = relevant, 1 = little relevant

They were weighted proportional to the item's importance for the problem's solution. 2 to 4 models per approach, which had the highest sum of weighted merit points, were chosen as "best referenced" for the practical qualification and final performance measure: 4 Lagrangian, 3 Eulerian, 2 Puff and 2 fluid dynamic models.

3) Practical qualification phase

Here practical experience and quantitative results were gained from the "best referenced" candidates for the final model performance evaluation and selection (phase 4).
For this purpose the models were applied to the analysis of SIESTA, an international mesoscale transport experiment (Gassmann, 1987), which simulates the transport of toxic gases and aerosols in complex terrain. Mean wind and turbulence were measured in conjunction with atmospheric dispersion of SF6-tracer during weak-wind situations over the Jura ridge and the hilly prealpine region. To limit the effort, the SIESTA analysis is restricted to two different episodes, experiment No. 4 on November 24 with a wind from NEE in non stable situation, and experiment No. 6 on November 30 with a weak wind from SWW to S in a stable situation. The two tests, therefore, have different topographies and stratifications. The tracer was sampled on 4/6 arcs at distances up to 35/25 km. 6 h averaged windfield and turbulence data from measurements were proposed for the analysis. They are not thought to be the best suited for a SIESTA based model validation study, as the experiment No. 4 shows. (There the calculated mean plume direction lies by some degrees outside the outer arcs in mountain regions, where arc measurements were forseen, but could not be completed due to difficult snow conditions.) However, this simplification should be sufficient for the relative comparison of the model candidates.
The menu driven program ANNE_LYSE was developed for real time analysis of the results produced by the different models. Multi criteria and different visualization techniques are used as iso-concentration lines (Fig. 1 and 2).

4) Final evaluation of model performance and selection phase

Here the "best" model candidate is evaluated and selected on the basis of the SIESTA analysis results and in respect of the performance criteria of phase 2. Based on model performance studies (Munn, 1987; Herrnberger, 1991) we decided in favour of a visualization and a statistical analysis of the results as for an intercomparison of the used cpu-time. The methods were built in the code ANNE_LYSE:

Visualization	Statist. analysis	Soft-/hardware performance
- Lateral distributions	- χ^2_2	- cpu time for same
- Scatter plot	- χ^2_{REL}	compiler optimization
- Cumul. frequency distribution	- C_{CORR}	hardware
	- Error factor Q	

Lateral concentration profiles per arc are visualized and position of center of concentration, maximum and horizontal width of the plume determined.
The cumulative frequency distribution indicates how many ratios of pairs of measured and calculational data (PMCD) or calculational and measured data (PCMD) > 1 lie within a certain factor (Fig. 3).
c^2: It is defined as the mean square deviation between measurements and calculations. Therefore the (absolute) importance of each PMCD is only proportional to its difference independent how large the data are. χ^2 can have any positive value. It is meaningful for relative comparisons of different methods with the same measurement.
χ^2_{REL}: To avoid the effect of predominance of high absolute differences and to weight equally each PMCD, the mean relative (standard) deviation is used, ranging inbetween 0 and 1. Small (Large) values mean good (bad) agreement between measurements and calculations.

C_{CORR}: This coefficient is some sort of a linearity measure inbetween measurements and calculations. If all PMCD are strongly correlated/anticorrelated, having the same linear dependence, C_{CORR} is about 1/-1. If the PMCD have different linearity or contain systematic or statistical errors $|C_{CORR}|$ is lower than 1! There is good agreement between calculated and measured data, if C_{CORR} is great and the proportionality factor is about 1.
Q: The mean error factor is related to the frequency distribution of the maximum ratios of PMCD. Symetric (normal e.g.) and narrow (delta e.g.) distributions correspond to values inbetween 1 (neglegible variation) and 2 (an error factor of 2). Asymetric or wide distributions have larger values.
To get relevant and comparable cpu times the models were compiled without optimization and run in scalar mode by the help of the modellers on comparable PSI hardware as VAX, CONVEX or IBM-RISC.

General conclusions

A) For dispersion modelling in complex terrain

- Transformation of windfields from terrain following to cartesian coordinates can influence dispersion, if topography forces the wind to produce bifurcation e.g.
- Interaction of upper with lower wind layers are not neglegible: Puff model show weak performance
- Planetary boundary layer stratification and turbulence description differ from model to model.

B) For model evaluation and selection procedure

- 4 phase evaluation technique is successfully applied to software packages of physical models of different complexity
- The screening process in 2 steps is efficient.
- Model performance measure and selection is guided by qualitative (visualized) and quantitative criteria.

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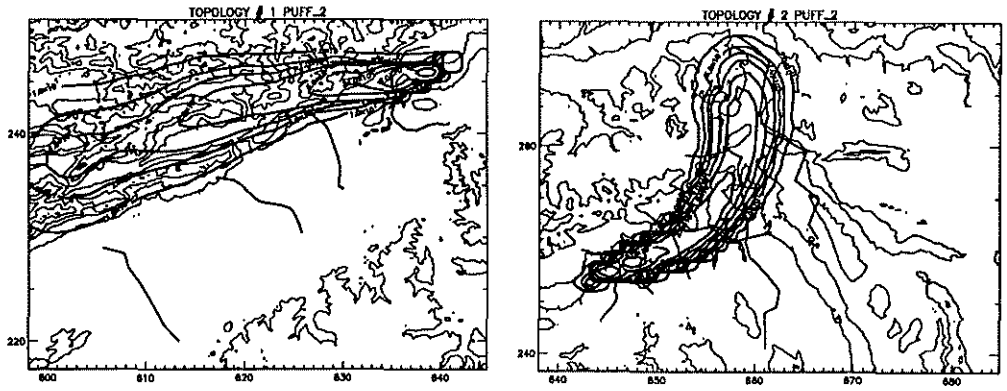


Fig 1: Iso-concentration lines of model PUFF_2 for SIESTA 24th & 30th

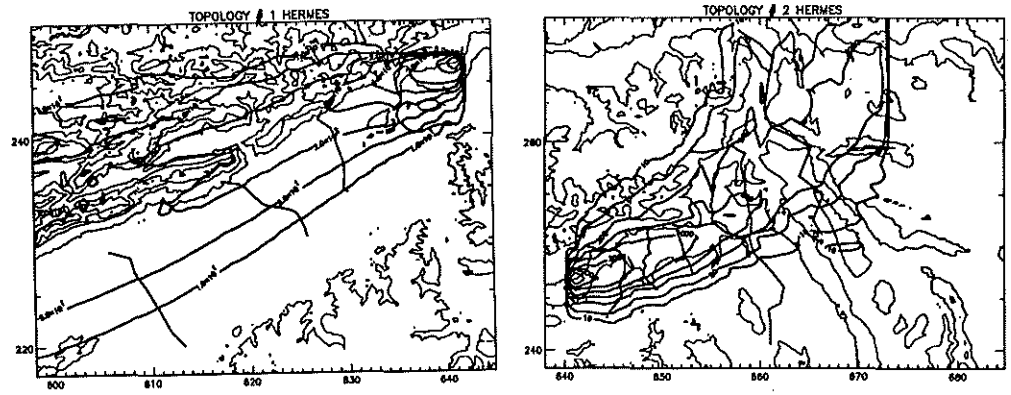


Fig 2: Iso-concentration lines of model Euler_2 for SIESTA 24th & 30th

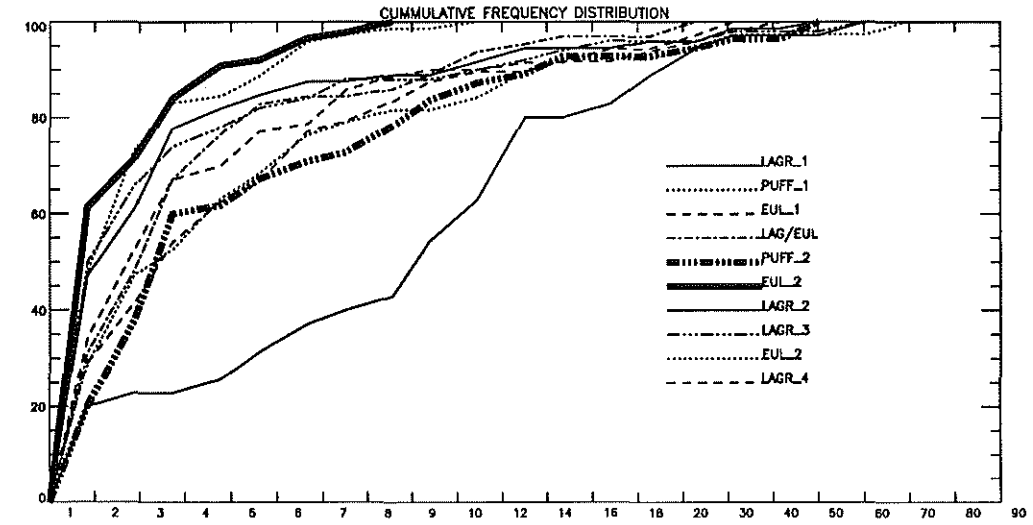


Fig. 3: Cumulative frequency distribution [%] of different models for SIESTA 30th versus PMCD- or PCMD-ratio > 1

PARAMETRISATION OF TURBULENT DIFFUSION IN A RIVER FROM SPEED FLUCTUATION ANALYSIS

Olivier MASSON

INSTITUT DE PROTECTION ET DE SURETÉ NUCLÉAIRE
Département de Protection de l'Environnement et des Installations
Service d'Etudes et de Recherches sur les transferts dans l'Environnement
IPSN/CEA, 13108 SAINT PAUL LEZ DURANCE CEDEX, France

Concentration distribution of a liquid waste, radioactive or chemical, in a stream is linked to both the emission features and to those of the flow regarding size, discharge and turbulent diffusivity.

In a given point and at a given time, heat and mass flux transfers are linked to the neighbouring environment and also to its history. This notion of flux is thereby dependent of a spatial area and a time area whose sizes can be approximated with the help of typical scales of turbulence. The study of this last notion looks essential since its a intrinsic component of all river and the instigator parameter of diffusion.

Thereby, the study of the dispersing power of a stream from a velocity area analysis seems to be the most physically appropriated method.

Analysis of velocity fluctuations, determined with the help of an ultrasonic curentmeter allows us to point out this turbulent diffusivity. Within the framework of research performed in an industrial canal and in a river wider than 150 m, it was possible to point out the size of the most powerful eddies in relation to the diffusion phenomena and the turbulence intensity.

Through the use of approximating relations which allows us to make the lagrangian parameters correspond to the measured eulerian parameters (β), a local parametrisation of turbulent diffusivity (K) was defined on an eddy length scale L_x and a longitudinal turbulent intensity $\sqrt{\overline{u'^2}}$, giving the following:

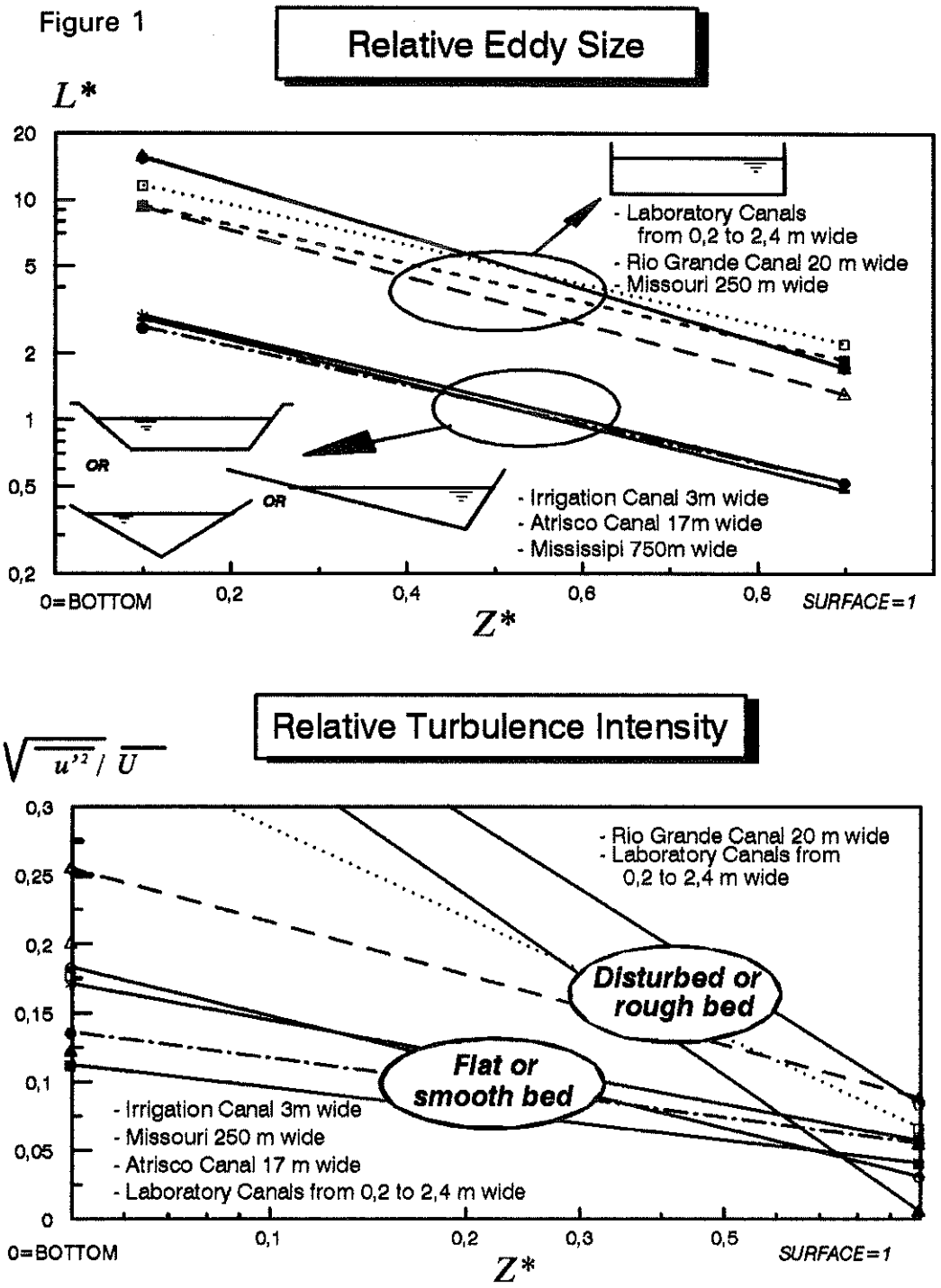
$$K = \beta . L_x . \overline{U} . \left(\sqrt{\overline{u'^2}} / \overline{U} \right)^2$$

The analysis of the terms of this relation has been the object of numerous comparisons with experimental data found in literature (Cf. figure 1) regarding the relative position above the bed level. More than 2500 values are synthesized on this figure including data from laboratory canal to the Mississipi river. Results of this comparison show that the data can be roughly divided in two categories, namely (1) the cross section shape and (2) the state of bed roughness.

A pragmatic use of those results aiming at forecasting the consequences of a liquid release into the environment led to a set of simplified "prescribed directions" based on the flow geometry and the river bed roughness which allows us to quickly estimate the terms of the proposed relation and to deduce values of diffusivity coefficients in the three directions.

Velocity measurements have been completed with concentration measurements of a tracer (Cf. figure 2). Those measurements have permitted us to follow space-time changes of the plume stream-wise and to deduce a macro turbulent mixing coefficient. The coefficients calculated by the two methods are in accordance (Cf. figure 3). The formula proposed is thereby validated and the work done gives a pragmatic tool for the help of diffusion coefficients determination used in diffusion and impact previsional models.

Figure 1



$\bar{U}$: Mean speed	Z : Height of measurement	L : Eddy size
H : Total height	Z* : Relative height= Z/H	u' : Fluctuations of U

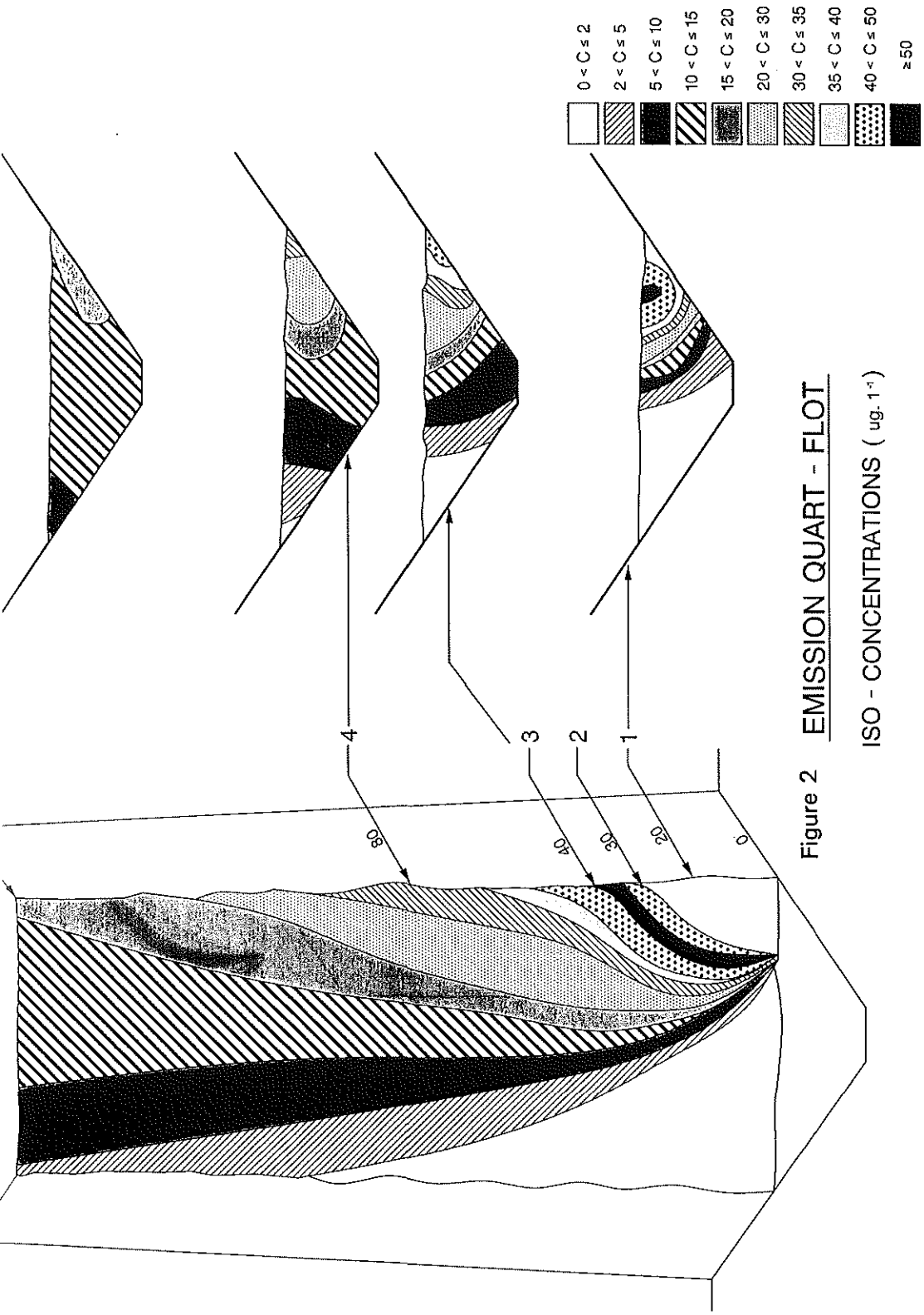
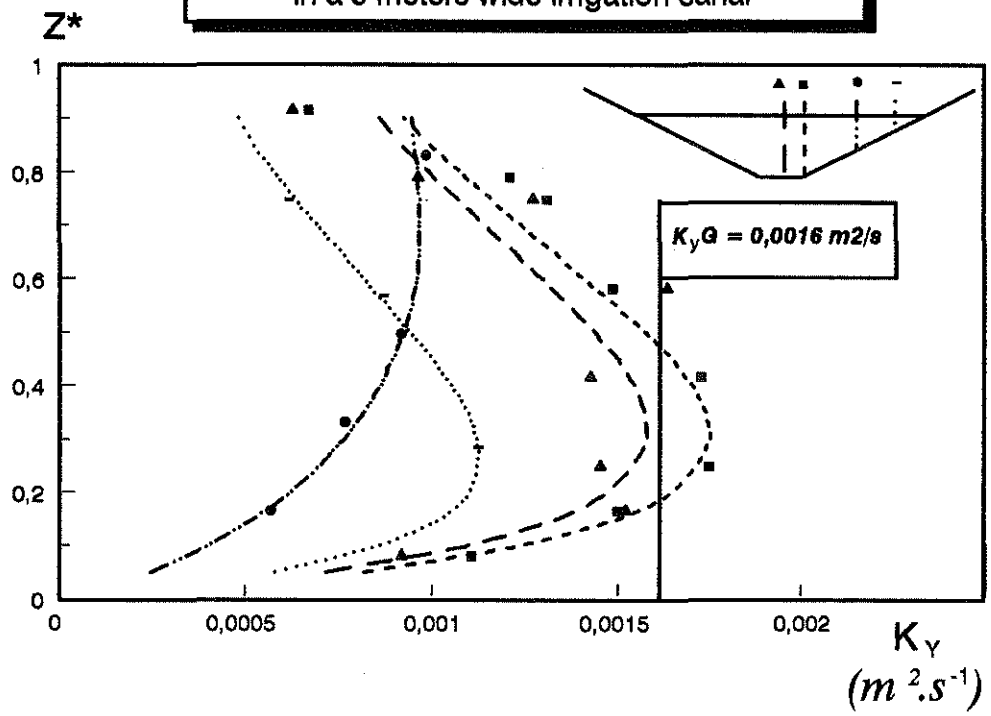


Figure 2 EMISSION QUART - FLOT

ISO - CONCENTRATIONS (ug.1^{-1})

Figure 3 Vertical profils of turbulent eddy coefficient in a 3 meters wide irrigation canal



Z* = Z/H

K_yG: : Global cross section diffusion coefficient from tracing experiment

K_y : Eddy cross section coefficient from velocity measurements

SODAR MEASUREMENTS AT AN OROGRAPHICALLY STRUCTURED SITE

E. Voelz, PreussenElektra AG, Hannover
H.J. Kirtzel, Metek GmbH, Hamburg

Summary

SODAR measurements performed at a German power plant were considered to be influenced by an effect that had simulated fixed echoes. In order to qualify the SODAR data, a mobile SODAR system was operated simultaneously with nearly identical operation parameters - only the transmit frequency was changed to prevent any interference between the systems. The intention was to investigate whether data quality could be improved by changing antenna alignments or operational frequency thus aiming for a set of more representative meteorological dispersion variables.

Our measurements showed that the influence of the fixed echoes was significantly reduced for the mobile system, when operated with identical antenna alignment but higher frequency than the stationary unit. The same result was obtained for a different alignment, but in this case no conclusion could be made whether alignment or frequency shift was responsible for the improved data quality.

Principles of SODAR technique

Since summer 1988 wind and turbulence measurements are performed at a site characterized by considerable orographic structures and large buildings in the surroundings. The data are used to calculate the dispersion of airborne materials. A conventional mast instrumentation is available for the 15 m height level and a SODAR system is used for the range from 50 m up to 350 m.

SODAR systems operate by transmitting sound pulses by three acoustic antennas that are orientated into different directions, and by receiving the backscattered signals from the atmosphere. By measurements of propagation time, signal intensity and frequency shift which is due to the Doppler effect of scattering processes in the moving air, height-dependent profiles of the horizontal wind vector and the vertical wind variance are deduced. Contrary to conventional in-situ measuring systems, SODAR techniques yield spatial and temporal averaged data. Important requirements to the site where a SODAR shall be positioned, are the horizontal homogeneity of the atmosphere and an avoidence of fixed echoes for all acoustic antennas.

Fixed echoes are recognized by a local maximum of signal intensity in the range of the transmit frequency, whereas the applicable atmospheric signal appears with the typical Doppler frequency shift proportional to the velocity of the scattering structures. In case of very low winds or for antennas orientated perpendicular to the wind stream a very small Doppler shift occurs. Therefore, both signals are mixed up preventing a reliable detection of and protection against fixed echoes.

In order to prove fixed echoes to be responsible for the unexplainable values at the height of 150 m, a second but mobile SODAR system was operated with identical operational parameters simultaneously to the stationary system at the same site. Only the transmit frequency was increased from 1675,6 Hz to 2200 Hz to avoid any interaction between both systems. For a first one-week-period the mobile antennas were aligned in the same way as the stationary antennas. For two more periods the mobile system was run with an alignment of the antenna A2 opposite to the direction of the corresponding stationary antenna. Both systems compare roughly in design and performance characteristics, but differ in the signal processing. In the stationary system spectral analysis is done by electronic filters whereas a Fourier transformation of the scattered signals is performed in the mobile system. It was proven that the different ways of analysis do not influence the derivation of mean wind values.

Results of comparison

As a consequence of the smaller size of the mobile antenna shields, an increased influence of echoes could be expected for this system. However, the applied analysis of the internal plausibility checks showed that the mobile system was hardly influenced by fixed echoes even when the antennas of both systems were identically aligned. The height-dependent wind velocity statistics of both systems are compared in Table 1 and 2.

It showed that the antenna beam pattern was considerably narrowed by increasing the transmit frequency just by some hundred hertz thus leading to a vanishing influence of the fixed echoes to the system. The same results were found during the other two periods when the antenna A2 was adjusted to the opposite direction, but in this case it could not be decided whether the differences in the antenna alignment or in the transmit frequency caused the improvement of data quality. Nevertheless, it can be concluded that increasing the transmit frequency may be an easy way to get rid of the influence of fixed echoes. That means that a SODAR system can be used for routine measurements at a structured site furthermore if only a fitted transmit frequency is used.

Site conditions

The site is situated in a turn of the Weser-river shielded to the south and west by a mountain ridge ascending steeply up to 150 m height and in a more gentle slope up to 250 m above the ground level. This ridge represents a very effective shielding of the local wind field from the predominating south-westerly wind stream. The measuring systems are located in the north-east 150 m far off large plant buildings at a height of 60 m and northwards to 40 m high cooling towers. Due to these special site conditions low wind velocities, deflection of wind directions and an increase of mechanically induced turbulence is expected.

Data of a comparable meteorological instrumentation located 40 km northwards in an orographically flatter region showed no discrepancy from the expected annual wind statistics.

Results of routine measurements

Routine data analysis of the SODAR measurements showed the following characteristics:

- Significant channel effects due to the mountain ridge are recognized at the lower heights. With increasing height, the distribution of wind direction compares better to the statistics of the geostrophic wind.

- The mean wind velocity is low at the 15 m level and increases slowly with increasing height.

- The stability classes that are derived by the vertical wind variance and the horizontal wind velocity and that are used as an estimate for the atmospheric turbulence, accumulate at labile and convective classes. This effect is only weakly reduced for upper height ranges. Stable stratification is hardly observed.

- All measured variables show a plausible tendency with height, but are disturbed at a measuring height of 150 m. Examples of a disturbed horizontal wind vector are shown in Fig. 1.

Influences due to the special orographic conditions at that site were expected. Nevertheless, the strength of the orographic effects was surprising. No reasonable explanation could be given for the disturbence of the measurements in the 150 m height. Therefore precise investigations were initiated.

Additional tests and measurements

In order to exclude any systematic technical failure of the hardware and software components of the system, the signal quality was tested by internal plausibility checks. The results showed that often secondary local maxima occured in the signal spectra for antenna A2, which typically indicates the influence of fixed echoes. Comparison with the results of the system 40 km apart gave no hints at a failure or malfunction of the system.

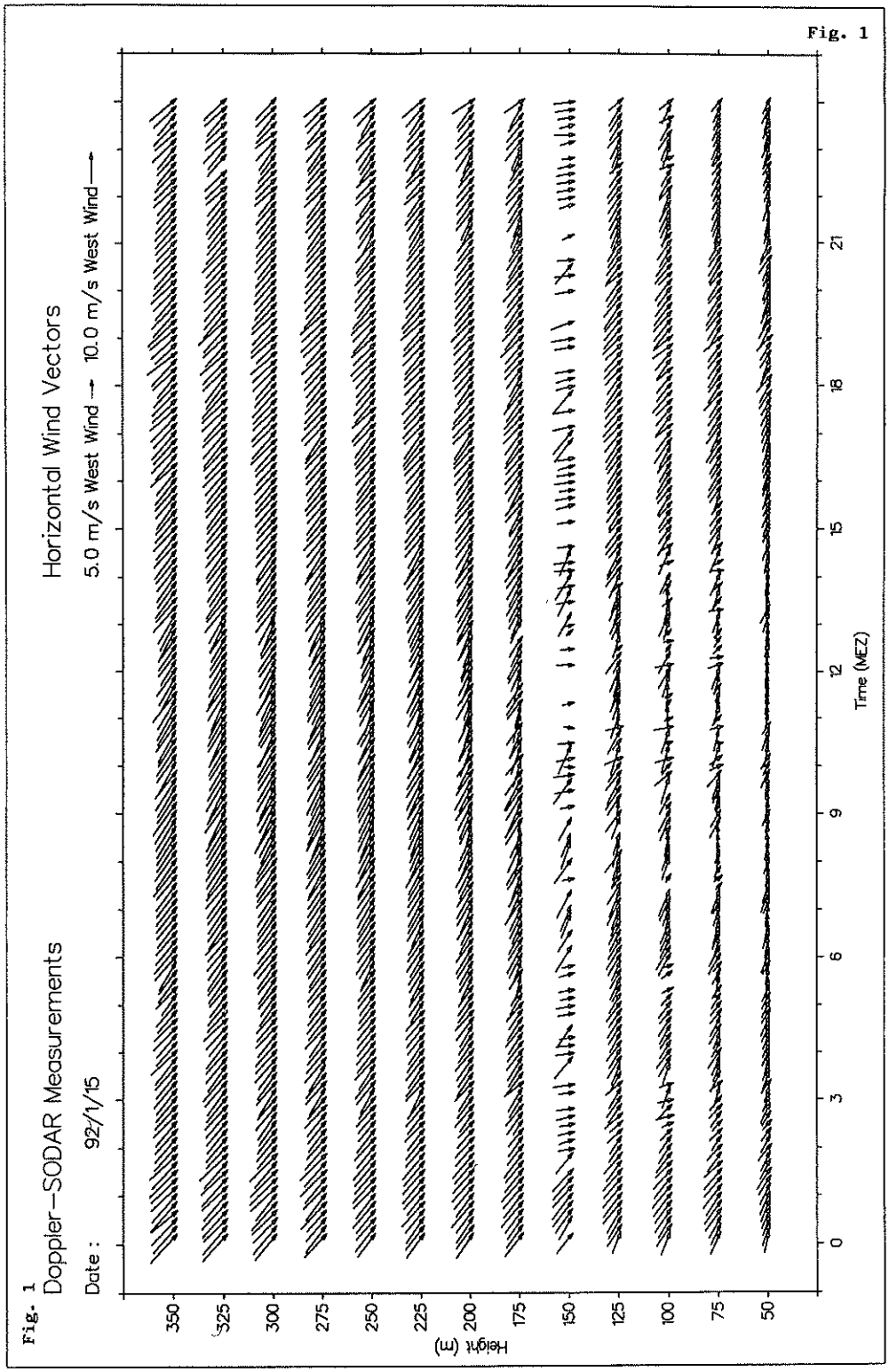


Fig. 1
Doppler–SODAR Measurements
Date : 92/1/15
Horizontal Wind Vectors
5.0 m/s West Wind → 10.0 m/s West Wind →
Height (m)
350
325
300
275
250
225
200
175
150
125
100
75
50
0
3
6
9
12
15
18
21
Time (MEZ)

Fig. 1

TABLE 1

Statistics of Variable V Doppler - SODAR DS100 (1675Hz)

Time Period : 23.10.91 13:45: 1 - 30.10.91 12:26:39 stationary system

Number of evaluated data sets : 1054

Height intervals :	50	75	100	125	150	175	200	225	250	275	300	325	350	total
Available data :	1047	1048	1028	1050	993	1040	1045	1044	1034	1036	1012	997	965	13339
Velocity	Number													
0dm/s - 15dm/s	222	249	268	173	303	124	119	70	76	57	34	23	17	1735
15dm/s - 30dm/s	139	102	88	95	77	99	86	86	57	50	51	43	36	1009
30dm/s - 45dm/s	203	149	131	123	79	81	67	79	76	61	41	39	31	1160
45dm/s - 60dm/s	356	300	256	289	211	210	153	123	121	118	107	88	67	2399
60dm/s - 75dm/s	110	208	222	277	224	312	255	220	169	180	191	165	171	2704
75dm/s - 90dm/s	12	27	47	65	59	132	208	241	226	220	211	214	206	1868
90dm/s - 105dm/s	5	12	11	22	31	65	132	178	223	237	248	255	226	1645
105dm/s - 120dm/s	0	1	5	6	6	13	21	37	73	92	87	107	123	571
120dm/s - 135dm/s	0	0	0	0	3	4	2	8	9	16	32	43	54	171
135dm/s - 150dm/s	0	0	0	0	0	0	2	2	4	5	10	20	34	77

TABLE 2

Statistics of Variable V Doppler - SODAR DSDS3x7 (2200HZ)

Time Period : 23.10.91 13:45: 1 - 30.10.91 12:26:39 mobile system

Number of evaluated data sets : 960

Height intervals :	50	75	100	125	150	175	200	225	250	275	300	325	350	total
Available data :	379	926	886	855	818	785	739	674	580	456	347	260	169	7874
Velocity	Number													
0dm/s - 15dm/s	66	132	126	102	124	66	44	29	22	18	9	6	1	745
15dm/s - 30dm/s	45	96	99	92	84	96	70	59	40	19	18	14	18	750
30dm/s - 45dm/s	43	160	151	117	79	64	81	66	56	42	23	12	1	895
45dm/s - 60dm/s	139	235	207	201	190	122	84	62	61	50	39	30	11	1431
60dm/s - 75dm/s	42	211	198	210	174	192	145	84	57	60	63	49	23	1508
75dm/s - 90dm/s	24	76	85	100	122	171	171	183	130	91	78	54	45	1330
90dm/s - 105dm/s	19	13	13	23	35	57	109	121	140	113	69	57	33	802
105dm/s - 120dm/s	1	2	6	4	5	13	22	53	56	44	35	28	28	297
120dm/s - 135dm/s	0	0	1	4	3	3	9	11	14	13	10	6	3	77
135dm/s - 150dm/s	0	1	0	2	2	1	4	6	4	6	3	4	6	39

IN SITU EFFLUENTS DISPERSION SIMULATION IN THE SEA USING A COLOURED TRACER

Gérard DEVILLE-CAVELIN

Institut de Protection et de Sûreté Nucléaire
Département de Protection de l'Environnement et des Installations
Service d'Etudes et de Recherches sur les transferts dans l'Environnement

IPSN/CEA 13108 SAINT PAUL LEZ DURANCE (FRANCE)

1 Aim of the experimentations

The aim of the experimentation was to know the dispersion characteristics of liquid plume in the sea close to the possible release point of the planned nuclear power plant in Morocco on the site of SIDI BOULBRA. The chosen location is situated on the Atlantic coast half-way between SAFI and ESSAOUIRA (cf enclosed map).

This experimentation was performed as part of a contract between IPSN/DPEI/SERE and SOFRATOME at ONE's[1] request

2 Methodology:

The experimentation was performed:

- by simulating the releases with a coloured tracer (rhodamine B) at 2 km of the site. The emission was almost a pin-point and instantaneous one.

- then by determining the concentration field through direct measurements in sea water using spectrocolorimeters shipped on dinghies. Streamwise and spanwise profiles were so determined at different times after the release. Vertical measurements were performed also up to 4 m deep in the liquid plume at its presumed centre.

The locations of the boats at the measurement points were done using laser pointing system (or theodolites and triangulations) from the coast.

3 Results:

Results are here simply given as the variation of the plume width and length as functions of time. Those dimensions are, first, compared with the forcasts of the DOURY-BADIE model[2].

3.1 Horizontal dispersion

3.1.1 Plume width :

The experimental curve is drawn following the equation:

$$width = (A.t)^k$$

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(1)Office National de l'Electricité du Maroc (NATIONAL ELECTRICITY BOARD OF MOROCCO)

(2)A. DOURY et C. BADIE - Une méthode pratique pour la prévision numérique des pollutions océaniques - Rapport CEA-R-4512 - 1973

These experimental values are compared with the width $(-3\sigma, +3\sigma)$ given by the model.

With the object of taking into account the fact that the emission has not a pin-point localization, a "virtual emission instant" has been introduced (-50 mn) in the model for calculating σ_y in order to give the width really measured at the instant of the first measurement (8 mn).

In the horizontal plane DOURY and BADIE proposed a variation law of the following form:

$$\sigma_y = (A_y . t)^{k_y}$$

with:

$A_y = 4{,}87 . 10^{-4} (in \quad m^{1/1{,}17} . s^{-1})$

$k_y = 1{,}17$

$t \quad in \quad seconds$

$\sigma_y \quad in \quad meters$

It appears (Fig.1) that for this experimentation, the agreement between experimental values and the model used is satisfying.

3.1.2 Plume length:

In the same way, measurement results for the plume length are compared with forcasts of the DOURY-BADIE model (fig.2). First, the evolution of the length $(6 . \sigma_x)$ does not seem to follow a law of the expected form $\sigma_x = (A_x . t)^{k_x}$ remaining the same during the whole duration of the experimentation (3 hours).

The coefficients

$A_x = 4{,}87 . 10^{-4} (in \quad m^{1/1{,}17} . s^{-1})$

$k_x = 1{,}17$

$t \quad in \quad seconds$

$\sigma_x \quad in \quad meters$

proposed in the model give values lower by a factor of 2 than the experimental ones at the beginning of the variation curve, and, three hours after the emission, by an almost 3 factor.

However, up to almost 70 minutes, the evolution law for the length can be estimated with the same kind of formulae than that proposed by DOURY and BADIE and using the coefficients:

$A_x = 2 . 10^{-4} (in \quad m^{1/1{,}36} . s^{-1})$

$k_x = 1{,}36$

and, after an hour, however with a worse agreement, we have found the following adjustment, of the same form as that previously mentioned:

$A_x = 1{,}8 . 10^{-6} (in \quad m^{1/1{,}9} . s^{-1})$

$k_x = 1{,}9$

3.1.3 Conclusions for the horizontal dispersion

The dispersion is anisotropic in the horizontal plane, and we have:

$\sigma_h = \sqrt{\sigma_x . \sigma_y}$

which could be written in the form:

$\sigma_h = (A_h . t)^{k_h}$

with:

* up to 70 minutes

$A_h = 3{,}52 . 10^{-5} . m^{1/1{,}27} . s^{-1}$

$k_h = 1{,}27$

* after 70 minutes

$A_h = 4{,}02 . 10^{-8} . m^{1/1{,}54} . s^{-1}$

$k_h = 1{,}54$

3.2 Vertical dispersion

Vertical measurements were restricted to a few meters below the sea surface because of very difficult experimental conditions (heavy swell).

However it appears that vertical profiles of the concentration at the presumed centre of the plume show an appreciable gradient at the beginning of the experimentation and become uniform 2 hours after the emission.

4 Conclusions:

- in the horizontal plane, dispersion is anisotropic in the studied zone,
  - for the streamwise dispersion, the coefficients proposed by DOURY and BADIE following upon OKUBO[3] seem to be in good agreement with the experimental results,
  - for the spanwise dispersion, the diffusion coefficient is very much higher than the streamwise one.
- vertical concentrations tend to become uniform during the first hours.

(3)A. OKUBO - A new set for oceanic diffusion diagrams - Technical report 38 - Chesapeak Bay Institute - The John Hopkins University - June 196

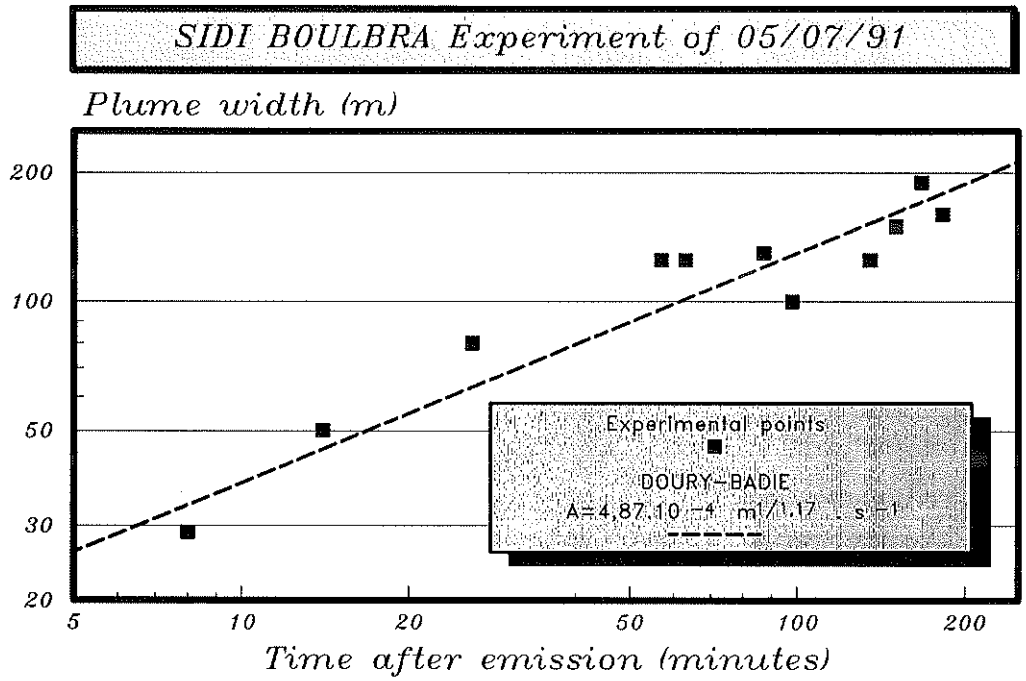


Fig.1: Variation of the plume width

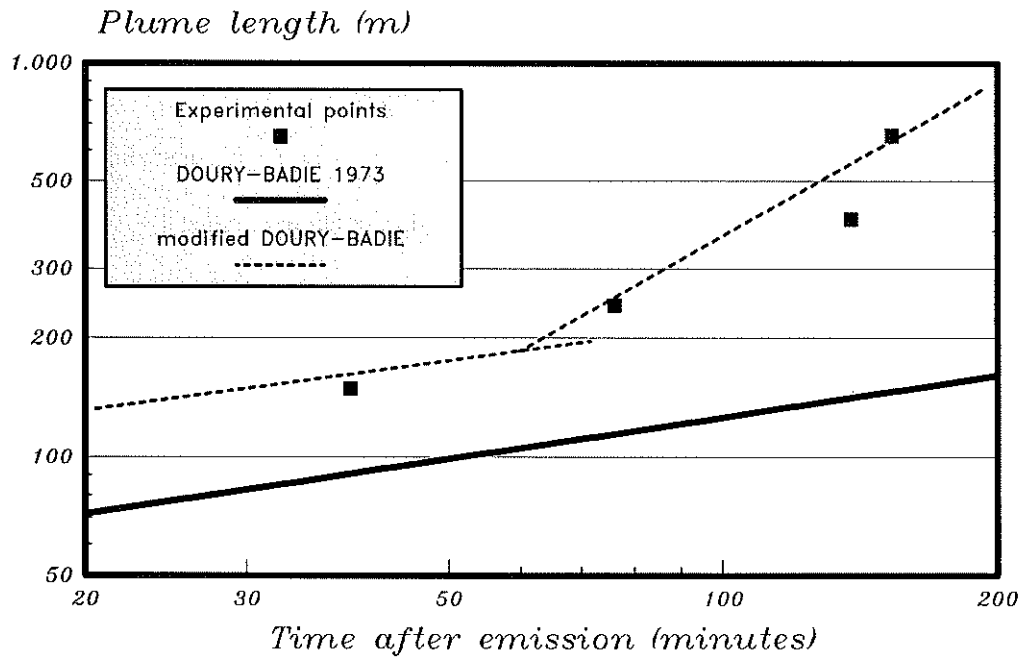
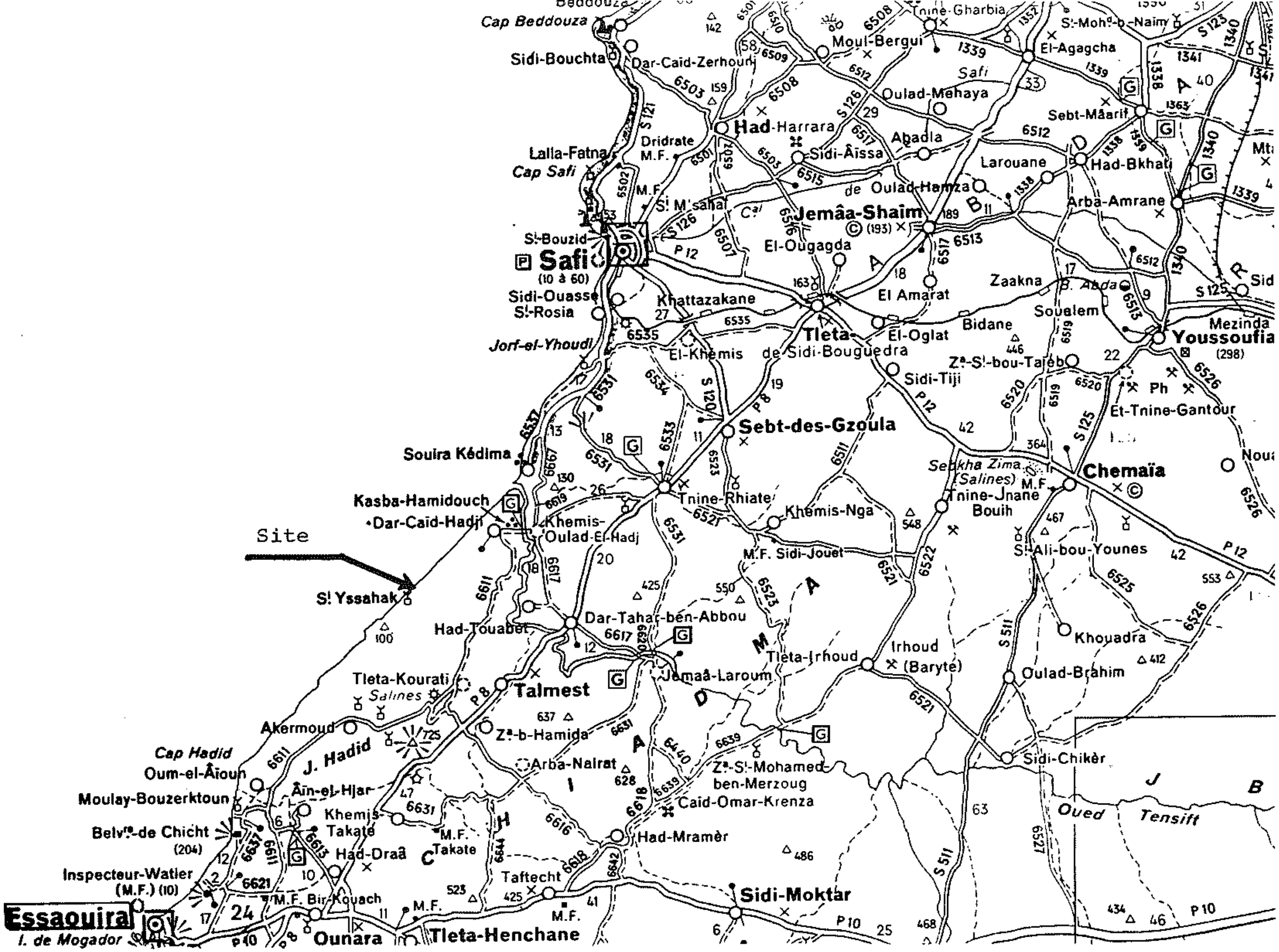


Fig.2: Variation of the plume length


Site
Cap Beddouza
Sidi-Bouchta
Dar-Caïd-Zerhouni
Moul-Bergui
Tnine-Gharbia
El-Agagcha
Safi
Oulad-Mehaya
Sebt-Mâarif
Had-Harrara
Abadla
Sidi-Âissa
Dridrate
M.F.
Lalla-Fatna
Cap Safi
Larouane
Had-Bkhati
Oulad-Hamza
Arba-Amrane
S! M'sahal
Jemâa-Shaim
El-Ougagda
S!-Bouzid
Safi
(10 à 60)
Sidi-Ouasse
S!-Rosia
Khattazakane
El Amarat
Zaakna
Tleta-de-Sidi-Bouguedra
El-Oglat
Bidane
Soualem
Mezinda
Youssoufia
(298)
Jorf-el-Yhoudi
El-Khemis
Sidi-Tiji
Z!-S!-bou-Tayeb
Ph
Et-Tnine-Gantour
Sebt-des-Gzoula
Souira Kédima
Sebkha Zima
(Salines)
Chemaïa
Kasba-Hamidouch
Dar-Caïd-Hadji
Khemis-Oulad-El-Hadj
Tnine-Rhiate
Khemis-Nga
Tnine-Jnane Bouih
S!-Ali-bou-Younes
M.F. Sidi-Jouet
S! Yssahak
Dar-Tahar-ben-Abbou
Had-Touabet
Khouadra
Tleta-Irhoud
Irhoud
(Baryte)
Jemaâ-Laroum
Oulad-Brahim
Tleta-Kourati
Salines
Talmest
Akermoud
Z!-b-Hamida
J. Hadid
Cap Hadid
Oum-el-Âioun
Arba-Nairat
Z!-S!-Mohamed-ben-Merzoug
Sidi-Chikèr
Moulay-Bouzerktoun
Aïn-el-Hjar
Caïd-Omar-Krenza
Oued Tensift
Belv!-de Chicht
(204)
Khemis-Takate
M.F. Takate
Had-Draâ
Had-Mramèr
Inspecteur-Watier
(M.F.) (10)
Taftecht
Sidi-Moktar
M.F. Bir-Kouach
Essaouira
I. de Mogador
Ounara
Tleta-Henchane

RADNUK - A DATA BASE AND INFORMATION SYSTEM ON RADIOLOGICAL DATA OF RADIONUCLIDES

K.–H. FOLKERTS, TH. SCHMITT, M. KELLER, M. SANDER

Hochschule für Technik und Wirtschaft des Saarlandes, FB Grundlagen, Informatik, Strahlenschutz, Goebenstr. 40, 6600 Saarbrücken/Deutschland

ABSTRACT

For a quick judgement of the radiological consequences by accidental releases of radionuclides a data base system for IBM compatible computers was developed. For more than 70 radionuclides from the reactor inventory as well as for some important natural radioisotopes relevant physical, radiation protection, physiological and biological data were entered. Physical data refer to the decay scheme, half lifes, emission probabilities and energies. Radiation protection data include informations on dose rate constants, dose factors for external and internal exposure by submersion, ground contamination and age dependent dose factors for inhalation and ingestion of radionuclides for the 5 most critical organs and the whole body. Physiological data refer to metabolic data, retention functions and general data as breathing rates and weights of organs. Applications are given for instance in civil protection for quick estimates of radiation exposures by accidental releases of radioisotopes.

INTRODUCTION

A comprehensive judgement of the radiological impact of accidental and routine releases of radionuclides for the general population needs many different data from different sources (1,2,3,4,5,6). Especially in case of accidental releases were quick and far reaching decisions have to be taken (Release of contaminated food, evacuation of parts of the population) looking for the necessary data from different sources may take valuable time. In order to sum up all necessary data in one source and to support a quick access to the relevant data the data base system RADNUK was developed.

DEVELOPMENT CRITERIA

The main aim of the work was the development of a comprehensive system running on an IBM compatible personal computer. (This type of PC's are widely distributed, so the software can be used without any additional hardware.) The available data base system contains up to now data on more than 70 radionuclides from the reactor inventory corresponding to the IMIS–list (IMIS : Integriertes Meß– und Informationssystem) and data on some important natural radionuclides. The data refer to physical, radiation protection and physiological/biological informations. Figure 1 shows the structure of the data in RADNUK.

PHYSICAL DATA

In this group, for every nuclide a simplified decay scheme, the physical half life, five decay energies for every kind of decay and emission propabilities can be found. Figure 2 shows a screen mask of RADNUK with the physical data of ^{131}J.

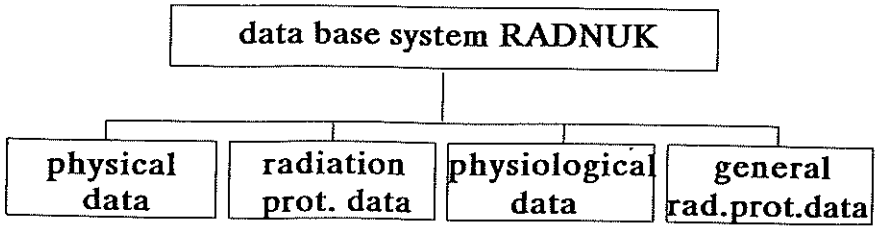


Fig.1: Data groups in the data base system RADNUK

```
HTW - FB GIS      R A D N U K     NUKLID - DATENBANK      Datum :   13.08.92
E X P E R T E  |  AUSWAHL : J -131 .. 340........J  |  P H Y S I K A L I S C H
NUKLID  : J -131 |  J -131  ==>1.1100% |100.00% ß- |      %    ==> Xe-131m
IMIS-NR :  340   |          ==>98.890% |100.00% ß- |      %    ==> Xe-131

Halbwertszeit  : 8.021E+00  [d]    Mutter   :          HWZ :            [d]
ZerfallsKonst. : 8.64E-02   [1/d]  Tochter1 : Xe-131m  HWZ : 1.184E+01  [d]
                                   Tochter2 : Xe-131   HWZ : Stabil     [d]

Emissionsenergien und -wahrscheinlichkeiten

      Eα [KeV]    Pα [%]      Eß [KeV]    Pß [%]      Eγ [KeV]    Pγ [%]
1)                            607         86.000      364         81.600
2)                            336         13.000      637         7.1200
3)                                                    284         6.2000
4)                                                    80          2.6800
5)                                                    723         1.7800

1 HILFE 2       3       4Strahl.5E_Spez 6       7       8       9       10ENDE
```

Fig.2: Screen display for physical data for ^{131}J.

RADIATION PROTECTION DATA

In this group information can be found on: dose rate constants, dose conversion factors for extern exposure by surface contamination and submersion, age dependent dose factors for inhalation and ingestion. Figure 3 shows the screen display for the radiation protection data of ^{131}J.

ELEMENT SPECIFIC DATA

In this part of the data base system the retention function, metabolic data and typical chemical combinations of the element can be found. These data are helpful for the calculation of long term exposure by incorporation of radionuclides and to determine the body compartments in which the element is mainly distributed.

GENERAL PHYSIOLOGICAL DATA

In this data group are stored: breathing rates, dose limits (whole body, surface contamination, contamination of food), mean food consumption, weighting factors w_T for the calculation of the effective dose.

```
HTW - FB GIS     R A D N U K     NUKLID - DATENBANK     Datum :   13.08.92
E X P E R T E  |  AUSWAHL : J -131 .. 340.....J  |  S T R A H L E N SCHUTZ

NUKLID  : J -131 |  J -131  ==>1.1100% |100.00% ß- .|     %    ==> Xe-131m
IMIS-NR :  340   |          ==>98.890% |100.00% ß-  |     %    ==> Xe-131

 Freigrenze :5.00E+04 [Bq]                    Dosisfaktoren Extern
 Dos.Le.Kon.:5.98E-11 [mSv*m²/Bq*h] Fl.b. 0.3g/cm²:1.27E-06 [(µSv/h)/(Bq/m²)]
 Max.Zu.Inh.:1.00E+06 [Bq]          Fl.b. 1.0g/cm²:1.06E-06 [(µSv/h)/(Bq/m²)]
 Max.Zu.Ing.:7.00E+05 [Bq]          Submersion ß  :5.28E-05 [(µSv/h)/(Bq/m^3)]
 Max.Ko.Luft:         [Bq/m^3]      Submersion γ  :1.02E-04 [(µSv/h)/(Bq/m^3)]
          Dosisfaktoren Intern [Sv/Bq]   Alter :  Erwachsener
     Organ(Inhal)  Day       Week      Year    Organ(Ingest)   1          2
Resorption        1.00E+00                                    1.00E+00
 1)  SCHILDDRÜS   2.7E-07                        SCHILDDRÜS    4.3E-07
 2)  LUNGE        6.5E-10                        MAGEN         3.0E-10
 3)  THYMUS       1.9E-10                        THYMUS        2.8E-10
 4)  GEHIRN       7.5E-11                        GEHIRN        1.2E-10
 5)  MAGEN        7.4E-11                        BRUST         1.1E-10
 6)  Ganzkörper   8.1E-09                        Ganzkörper    1.3E-08
 7d-Folgedosis für Knochenmark bei Inhalation [Sv/Bq] : 3.60E-11
1 HILFE 2Alter  3Phkal. 4       5E_Spez 6       7       8       9       10ENDE
```

Fig.3: Screen display with radiation protection data for ^{131}J.

The sources of the data refer to the German "Strahlenschutzverordnung", ICRP 30 and related sources. (1,2,3,4,5,6)

Beside this data numerous options for printing, conversion of the data into ASCII–format, deleting, changing data, expansion to other nuclides were implemented. Especially the possibility for access to the data by external programs (user written special dose calculation programs) was planed. In order to support a flexible use of the data base system it is planed to add specific dose calculation procedures to the system. The system ist intented for use with task forces in civil protection, radiation protection authorities, in teaching and in nuclear industry.

SOFTWARE, HARDWARE

The surface of the data base system is written in CLIPPER 5.0 and the data are stored in dbase compatible data format and are directly accessable by the external user. As hardware configuration a IBM AT compatible computer with at least 640 kbyte memory and 2 Mbyte of free hard disk space are necessary.

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ON THE LIBERATION AND SPREAD OF GASES EMANATING FROM THE GRANITIC HOST ROCK OF A HYPOTHETICAL REPOSITORY FOR RADIOACTIVE WASTES

Wolfgang Bode, Norbert Jockwer, Herbert Kull

GSF-Institut für Tieflagerung, Theodor-Heuss-Str.4, D - 3300 Braunschweig, Germany

Summary

In cooperation with Nagra* the GSF-Forschungszentrum für Umwelt und Gesundheit has been carrying out investigations in the rock laboratory Grimsel Test Site (GTS) since 1984 on the properties of crystalline rock formations in order to determine their suitability to host disposed radioactive wastes. In the current R&D-programme the influence of an evaporation zone and mineralogical structure on the fluid transport and on the water saturation in crystalline rock is studied. Besides in situ studies the thermal release and spreading mechanisms of natural gases in the evaporation zone are investigated in laboratory tests. Natural gas components are normally weakly bound on grain surfaces (e.g. methane, carbon dioxide and hydrogen) or part of the minerals (e.g. hydrocarbons). Intergranular pore spaces are considered to be the major pathways in the low-deformed crystalline rock. Further laboratory studies will examine the exchange of these natural gas components by technical waste bound gas components.

* **Na**tionale Genossenschaft für die Lagerung radioaktiver Abfälle

Introduction

Concepts for the disposal of radioactive wastes in geological formations require extensive knowledge of the proposed host rock and the surrounding formations. For this reason Nagra has been operating the Grimsel Test Site (GTS) since May 1984. On the basis of a German-Swiss cooperative agreement, various experiments are carried out by Nagra, the Bundesanstalt für Geowissenschaften und Rohstoffe, Hannover (BGR), and the GSF-Forschungszentrum für Umwelt und Gesundheit, GmbH, München (GSF). The projects of the German partners are supported by the German Federal Ministry for Research and Technology (BMFT).

The subject of the completed gas studies with core samples from the GTS was to investigate the natural gas components of the rock and to analyze their behaviour under thermal strain.

By emplacing radioactive waste in geologic strata, gases, which are absorbed or adsorbed to the host, rock will be released as a result of elevated temperature and microfissures generated by mining activities. Thermal and radiolytic decomposition of unstable components within the host rock will generate and release further gases.

These gases and volatile components in combination with the humidity within the host rock may lead to corrosion of the containment and of the solidification matrix which mobilizes the radionuclides. This process generates additional gas - especially hydrogen. If there is no drain for the hydrogen via diffusion or migration into the backfill material or into the host rock, a combustible atmosphere will accumulate within some years in the disposal area. Furthermore all these gases may be transport media for mobilized radionuclides.

Laboratory investigations

The GTS is located at a depth of 450 m in the crystalline rock of the Aare Massif of the Central Swiss Alps. The site-specific host rock is a metamorph affected magmatic intrusion of Grimsel granodiorite.

The laboratory studies were performed on core samples taken from the ventilation drift at GTS. The diameter of the cores varies between 50 and 250 mm, corresponding to a core length up to 250 mm. The cores were wet-drilled from the bottom of the drift. The orientation of boreholes was parallel to cleavage of the crystalline rock.

Mineralogy of crystalline rock

The drill cores were characterized by analyzing the feature's rock mineralogy by microscopic description, and the pore space using the impregnation technique with fluorescent resin.

The core samples are granodiorite gneisses. Figure 1 illustres the mineralogical composition of the samples in the STRECKEISEN-diagram with the major components quartz, plagioclase and potassium feldspar, but without biotite. Minor components are muscovite and epidote/clinozoicite. Accessory minerals are chlorite, titanite, zircon, apatite and opaque minerals.

The rock is of a medium grey colour, and the grain size is medium to coarse with porphyric blasts. Most of the hypidiomorph to idiomorph feldspar blasts are of dimensions up to 12*20 mm. Quartz minerals are generally small. Single xenomorph quartz grains of a size of between 4 to 8 mm showed intragranular pores.
The cleavage of this part of the rock - illustred by the deformation of biotite and muscovite-minerals - is weak to intensive. Typical sigmoidal cleavage structures change with small-scaled areas of relictic magmatic textures.

The porosity of the crystalline rock depends upon the intergranular pore space between the major minerals. The percentage on pore space of fissures in porphyric blasts (intragranular pore space) as well as pore spaces caused be solubility effects and open sheet-structures in biotite/muscovite minerals is negligible.

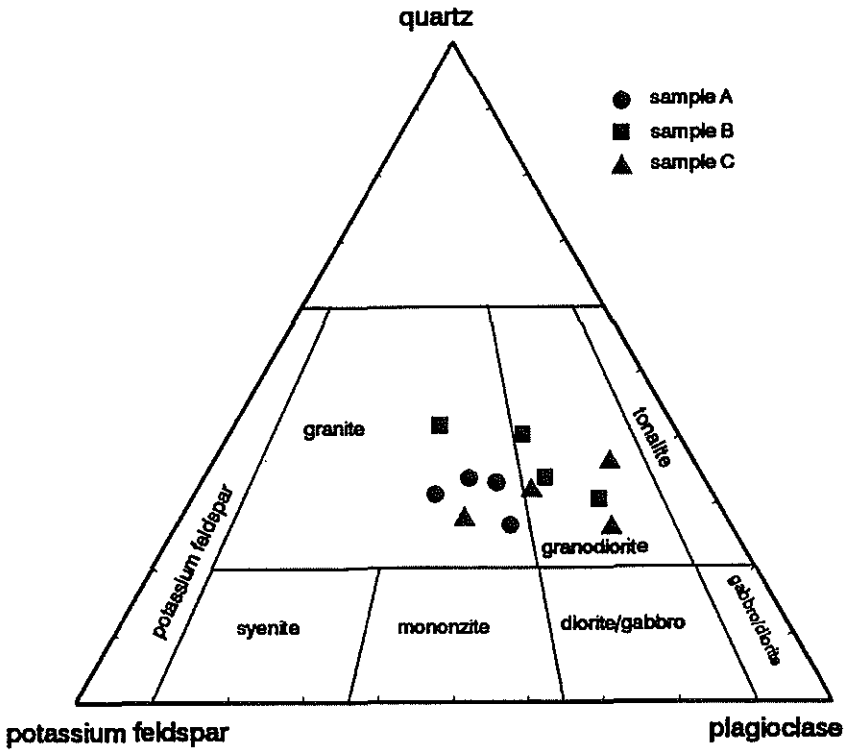


Figure 1: Major mineral composition of crystalline rock from the ventilation drift (GTS)

Natural gas components in crystalline rock

When developing a repository for radioactive waste all sources of gas generation and release, both during the operational phase and in the post-operational phase, as well as the behaviour of these gases in the backfilling material and in the host rock have to be known qualitatively and quantitatively.

For this purpose drilling cores from the Grimsel site with a diameter of 50 mm were taken. The first aim was to determine the primary gas content, its thermal release behaviour, and the thermal generation of further gases. Therefore the cores were ground to a grain size of 1 to 3 mm. About 3 g were heated in the oven of a mass-spectrometer, beginning at room temperature (about 20 °C) up to 500 °C at a heating velocity of 10 °C/min. Due to heating gases primarily present in the granite are released, but further gases are generated and released by thermal decomposition. All of these gases are analyzed qualitatively with the mass-spectrometer. The disadvantage of this method is that it is impossible to distinguish between gases already present in the host rock and those which are generated by thermal decomposition.

Figure 2 shows the results of the thermal gas release out of ground, pre-dried granite/granodiorite versus temperature determined using the mass-spectrometer. It indicates that granite/granodiorite contains, besides the adsorbed and included water, hydrogen, methane, hydrochloric acid and carbon dioxide. The release of water and methane begins at a temperature above 50 °C and it increases with increasing temperature. Hydrogen, hydrochloric acid, and carbon dioxide will be released at a temperature above 300 °C. All gases and the adsorbed water have their maximal release rate within a range between 400 and 500 °C.

The granite/granodiorite samples were dried at room temperature (20 °C) for some months and have a content of

water	in the range of	0.5	to	2	weight %
methane		50	to	500	ppm
hydrochloric acid			10		ppm
carbon dioxid		100	to	2000	ppm
hydrogen		10	to	100	ppm

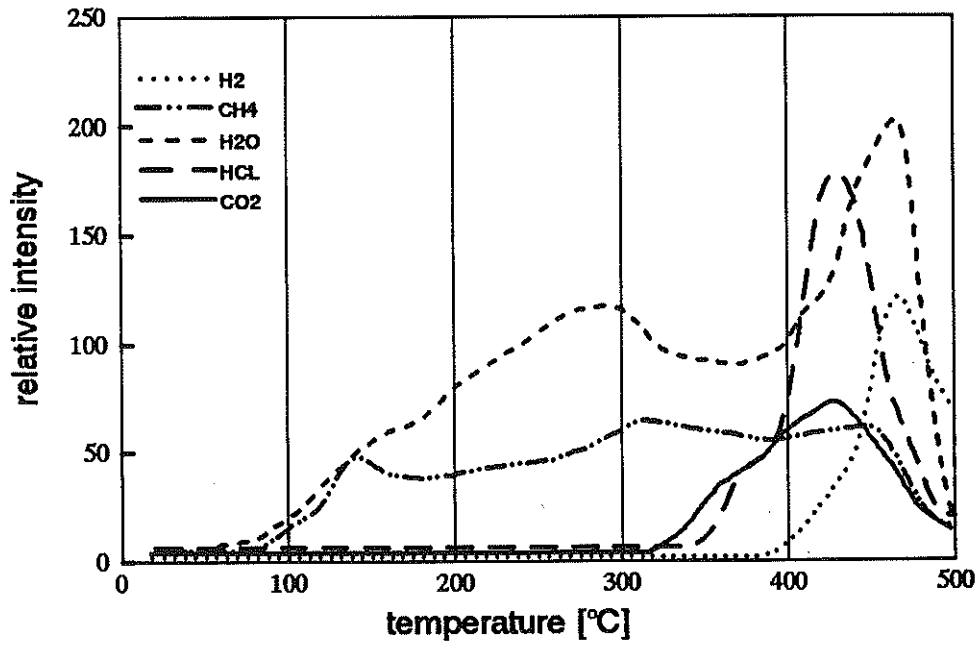


Figure 2: Thermal release of the gas components hydrogen,methane, hydrochloric acid, carbon dioxide and water from the granodiorite from the GTS

The samples which were pre-dried at room temperature are not representative for the undisturbed area of the host rock. However, they represent the dried zone around the ventilation drift or boreholes (evaporation zone).

The water and gas content of the granite/granodiorite in the uninfluenced zone is assumed to be significantly higher, but, as it is very difficult to get these samples into the laboratory, investigations on that subject have not yet been performed.

Petrophysical properties

One main subject concerning the released gases in a repository is their behaviour in the host rock, especially their migration and diffusion behaviour in combination with potentially released radionuclides. For this purpose the gas permeability of samples of a diameter of 50 mm and a length of 50 mm has been determined in a modified HASSLER-cell. The gas permeability of granite depends on the water content of the sample, the confining pressure, and the gas pressure.

At a confining pressure of 10 MPa the gas permeability of water-saturated core samples is within a range of 10^{-18} to 10^{-20} m^2. The gas permeability of pre-dried, water unsaturated samples increase to the range of 10^{-15} m^2.

The effective porosity of the low deformed granite/granodiorite is within the volumetric range of 0.5 to 1.5 %.

Conclusions

Natural gas components are analyzed in pre-dried core samples of granite/granodiorite from GTS. Water and methane are weakly bound to the minerals, carbon dioxide, hydrogen and hydrochloric acid are released by higher thermal strain. Further gas components are expected to be generated by radiation or corrosion. These gas components are the predominant transport media for released waste particles in the gas phase. Major flowpaths are the intergranular pore spaces of the low-permeable granite/granodiorite, assuming an evaporation of the crystalline rock. The interaction of natural and secondarily generated gas components with the crystalline rock will be analyzed.

Acknowledgment

The activities were carried out by the GSF - Institut für Tieflagerung (IfT) within the scope of a R+D-Project: "Untersuchung physikalischer Prozesse und Parameter zum Fluid- und Gastransport im Nahbereich von Endlagern in granitischen Formationen des Felslabors Grimsel (Phase 3)" - Förderkennzeichen: 02E8151 8 - supported by the Federal Ministry for Research and Technology (BMFT).

Part 3: Environmental Monitoring

ENVIRONMENTAL SURVEY AROUND *EDF* NUCLEAR POWER PLANTS

M. CARRE M. * - M. FOULQUIER L. ** - M. HARTMANN P. * - M. RUFENACH M. *

* *Electricité De France - Centre Nucléaire de Production d'Electricité de Fessenheim*
** *Institut de protection et de Sûreté Nucléaire - CEA*

SUMMARY

Description of various types of environmental test carried out under the responsibility of the Operator of nuclear power plants in France, with taking Fessenheim nuclear power plant as an example : permanent monitoring of radioactivity, periodic radioecological assessments, main results of measurements taken, showing that there are no detectable effects of the plant on the environment, policy of openness by publication of these results.

INTRODUCTION

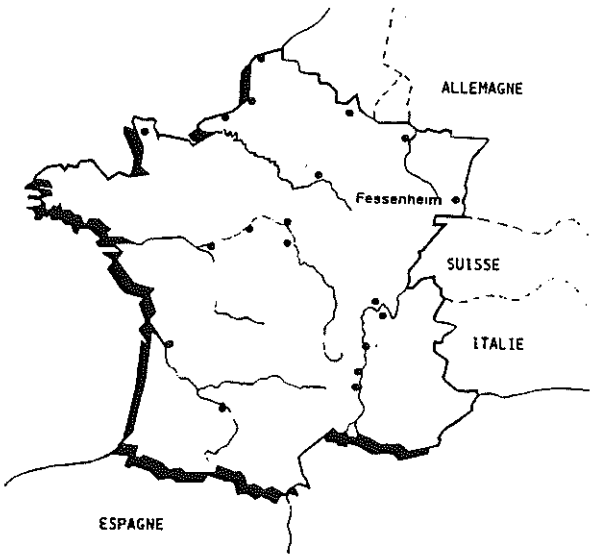
France has a large Nuclear Generating Capacity.

The environment must be a constant concern, renewed according to scientific and technological progress. Observing strict regulations, nuclear power plants release, after traitement, liquid and gas effluents, the radioactivity of which is added to radioactive fallout from old nuclear explosions and from still detectable effects of the Tchernobyl accident.

It is indispensable to monitor the surrounding radioecological state in order to ensure that operating these plants is compatible with conservation.

The Site of Fessenheim is given as an example of the various types of test carried out under the Operator's responsability.

1. FESSENHEIM NUCLEAR POWER PLANT



It is the oldest French 900 MW PWR plant.

It consists of two Units, connected to the grid in 1977.

In France, there are currently 54 Units distributed over 19 sites for 55 400 MW.

The general process for radioecological studies consists of 4 phases :

- initial radioecological assessment establishing a reference state before the plant commissionning,
- permanent monitoring of radioactivity in the area surrounding the plant,
- complete radioecological assessment made after approximately 10 years in operation,
- annual monitoring of main radio-indicators in the earth and aquatic ecosystems,

It is then possible to compare the results obtained with the effluent composition (table 1).

Table 1 : ***Overall composition of liquid and gaseous effluent released from Fessenheim nuclear power plant***

	SUM OF RE*(GBq)	TRITIUM (TBq)	GAS (TBq)	AEROSOLS + HALOGENS (GBq)
1985	82	30	93,5	0,12
1986	79	38	100	0,20
1987	42	36	44	0,11
1988	47	33	< 12	< 0,03
1989	46	24	< 11	< 0,03
1990	34	20	< 8,3	< 0,03
1991	18	26	< 14	< 0,04
**	925	75	1500	110

* Sum of radioelements apart from H3, Ra, K40.
These are basically Co 60 + 58 (about 80 % of release), Ag 110 m, Sb 124, I 131, Cs 137 + 134.

** Annual authorized limit for site.

2. INITIAL RADIOECOLOGICAL ASSESSMENT

It is obligatory for the study of environmental impact as one of the conditions of the Request to provide a Utility, before the first Unit is started up. It consist of two facets :

- Evaluation of dose, based on a series of measurements of dose equivalents in the region and analyses of γ spectrometry inside living quarters. Variations due to geology or manmade modifications are observed. In the Plain of Alsace, the results are from 0,66 to 0,96 mSv p.a with a supplementary contribution of between 0,26 and 0,61 mSv p.a, related to buildings. Occasional values are noted : 4,8 mSv p.a on a farm near a heap of manure and 33 mSv p.a near uranium-bearing lode.

- Measurements of radioactivity in the environment, mainly based, at this time, on soil, wines, corn, honey and tritium of water. The Plant has at its disposal data concerning Rhine fish. Their Cs 137 content, attributable to radioactive fallout from nuclear tests, varied from 1 to 6 Bq.kg-1 wet weight.

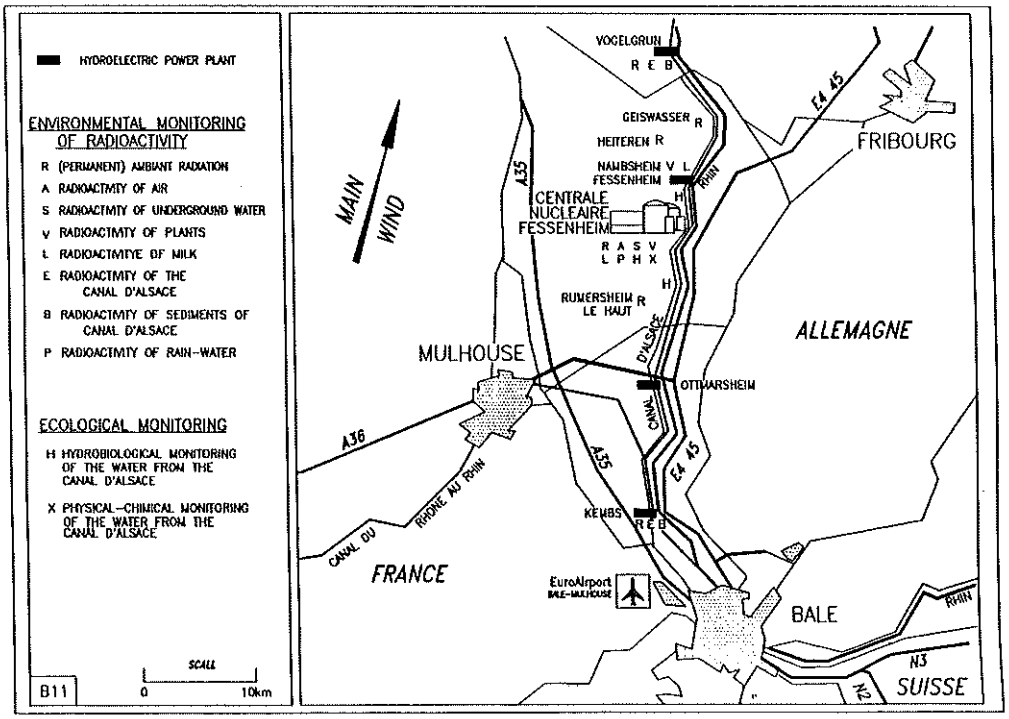
3. PERMANENT MONITORING OF RADIOACTIVITY

The measurements are carried out by the Operator (table 2) according to a statutory program controlled by the Central Service for Protection against Ionizing Radiation (SCPRI), which is responsible to the Ministry of Health : ③, ④, ⑤. The aim of this program is to detect any anomally by global measurement with daily, weekly and permanent monitoring. Every month, the plant distributes these results : to Local Public Authorities, elected representatives and media. The results are also displayed on Minitel (national public information service - MAGNUC). SCPRI carries out complementary tests concerning the whole of France, the results of which are also displayed on Minitel.

Table 2 : ***Measurements for monitoring radioactivity carried out by the Operator around Fessenheim nuclear power plant***

TYPE OF MEASUREMENT	AVERAGE RESULTS IN 1991	WHEN THE CLOUD FROM TCHERNOBYL WAS PASSING OVER (maximum values)
(Permanent) ambient radiation	0,01 to 0,04 μGy.h-1	0,15 μGy.h-1
Aerosols in air on filter, measured 6 days after sampling	totalβ : 0,34 to 0,70 mBq.m-3	totalβ : 12000 Bq.m-3
(Rhine) riverwater during effluent release	totalβ < 1,1 Bq/l	totalβ < 11 Bq/l
(Weekly) rainwater	totalβ < 0,61 Bq/l	totalβ < 62,3 Bq/l
(Monthly) groundwater	totalβ < 0,40 Bq/l	totalβ < 0,57 Bq/l
(Monthly) land vegetation	totalβ (excluding K40) = 160Bq.kg-1(d.w.)	
(Monthly) milk	totalβ (excluding K40) < 0,45 Bq/l	totalβ (excluding K40) : 290 Bq.l-1 I 131 : 140 Bq/l Cs 137 : 42 Bq/l

Main measurements of monitoring the radioactivity around Fessenheim nuclear power plant



4. TEN-YEARLY RADIOECOLOGICAL ASSESSMENT

A ten-yearly radioecological assessment is realised to improve the permanent radioactivity monitoring and find releases impact of the plant on the environment. This study ① was carried out after 12 years in operation, in Fessenheim, at the time of ten-yearly outage program for both units. It consists of spectrometry γ on all samples and measurements of H3, Sr and Pu on some of them.

Land ecosystem :

For the earth ecosystem, at stations representative of the compass, samples of soil, milk, irrigation water, produce (corn, wine, wheat, lucerne, ...), are taken and analyzed (table 3). The gaseous releases in atmosphere of the plant doesn't impact the land ecosystem. The arboreal mosses, excellent radio-indicators, contain radioelements attributable to radioactive fallout from Tchernobyl ; thorough analysis of the concentration and deposit of Cs 137 in the soil, taking into account the effect of radioactive fallout from Tchernobyl, has completely exonerated gaseous release from the plant.

Table 3 : ***Artificial γ radioactivity (in Bq.kg-1 dry weight) in some samples from the land around the plant in 1989***

	WINE SOIL	MOSS	LUCERNE	CORN
Cs 134	1,6 ± 0,5	81 ± 4	---	---
Cs 137	14 ± 2	413 ± 13	0,1 ± 0,1	0,19 ± 0,1
Ag 110 m	---	0,5 ± 0,3	---	---
Co 60	---	0,3 ± 0,2	---	---
Natural radioactivity	~ 1500	---	~ 2000	

Aquatic ecosystem :

The aquatic ecosystem presents some tiny presence of radioelements due to impact liquid effluent releases of the plant, but the impact health is not detectable. The results show that the Tchernobyl accident has greatly influenced radioactivity levels -water, sediment, vegetation, fish- (table 4). The presence of radiocobalt upstream from Fessenheim is attributable to effluent from a nuclear power plant on the upper Rhine. The isotopic ratio of C esium corresponds to that of Tchernobyl. Downstream, the concentration of Co 58 + 60 in aquatic vegetation can be attributed to effluent from Fessenheim nuclear power plant.

Table 4 : ***Examples of artificial γ radioactivity in the Rhine (in Bq.kg-1 d.w. for sediment and vegetation, in Bq.kg-1 w.w. for fish in 1989***

	UPSTREAM			DOWNSTREAM		
	Sediment	Vegetation	Fish	Sediment	Vegetation	Fish
Cs 134	18 ± 2	5,2 ± 1,9	0,4 ± 0,2	11 ± 3	2 ± 0,9	0,4 ± 0,1
Cs 137	97 ± 10	25,4 ± 2,8	2,2 ± 0,3	57 ± 3	7,5 ± 1,5	2,1 ± 0,2
Co 58	1 ± 0,5	8,5 ± 2,8	---	3 ± 1	69 ± 6	---
Co 60	4 ± 1	10 ± 3	---	3 ± 1	16 ± 2	---
Mn 54	1 ± 0,5	2,8 ± 1,4	---	---	3,3 ± 1,1	---
Natural Radioactivity	~ 1500	~ 1300	~ 400	~ 1500	~ 1300	~ 400

5. ANNUAL RADIOECOLOGICAL MONITORING

Electricité De France decided on an annual radioecological monitoring program for all Sites. These annual studies are realised to have a more regularly assessment of the environment impact ②. This program begun in 1992, around all the plant of EDF. Approximately thirty radio-indicators well known in land and water (continental and marine) can be monitored with γ spectrometry and, if necessary, radiochemical analyses. There will therefore be a data bank permanently updated, according to time and space.

CONCLUSION

The impact of 12 years operation of Fessenheim nuclear power plant is represented by a slight presence of artificial radioelements in the environment. These results underline the importance of radioecological testing before plant start-up and during its operation, thus enabling any anomaly to be discovered at any time. In accordance with a national desire for openness, the public is informed of the results of these tests by the authorities, elected representatives and media.

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④ Service MINITEL MAGNUC concernant l'environnement et la sûreté des installations

⑤ EDF DER
Revue hydroécologie appliquée n° 3 - Publication EDF

ENVIRONMENTAL MONITORING AROUND SWISS NUCLEAR POWER STATIONS (NPS)

CH. MURITH, H. VÖLKLE
Office fédéral de la santé publique, section de surveillance de la radioactivité
Chemin du Musée 3, CH-1700 Fribourg / Suisse

Abstract

In addition to the control and balance of radioactivity releases at NPS, an environmental monitoring programme is operated in the whole biosphere (air, water, soil, food, human body, etc.). This programme [1], set up in close cooperation with the laboratories concerned and the German authorities for the plants near the Swiss border (Leibstadt & Beznau), is based on the experiences obtained from the monitoring of radioactive fallout from nuclear bomb tests, on the knowledge of the behaviour of radioactive nuclides in the environment and on the composition of the releases of NPS. The sampling points have been chosen according to the meteorological observation programme operated at each site several years before the plant began to work. Sampling and measurement techniques are in accordance with internationally approved recommandations and with the actual scientific and technical knowledge. We present here the radioecological assessment of the 8 last years comparing artificial contributions (past atmospheric nuclear weapon test, Chernobyl, Swiss nuclear power stations) to natural radioactivity. Its results that the impact of Swiss nuclear power stations in the environment is very low and has never exceeded the exposure limit fixed by the authorities in such a way that no person of the surrounding population could in any way accumulate an additional dose of more than 0.2 mSv/y.

1. Exposure rate measurements

In addition to the automatic exposure rate surveillance network for the whole country (NADAM) and for the vicinity of nuclear stations (MADUK, installation on the way), ionization chambers (RSS) are used to register the exposure rate continuously in the two main wind directions and TL-Dosemeters to integrate the exposure quarterly. The detection limit obtained by an appropiated numerical evaluation technique [1] allows us to detect exposure contributions from the stack releases as low as some 20 - 40 μSv/y in comparison to about 1000 μSv/y from natural radiation. Especially in the case of BWR, another source of external radiation at some places directly outside the fence is the so called direct radiation of Nitrogen-16 (γ-rays up to 7 MeV) produced by neutron capture in the reactor. This contribution up to 5 mSv/y at the fence of the Mühleberg plant diminishes at a distance of 400 m to 0.2 mSv/y and is less than 20 μSv/y at the nearest inhabited houses. Fig. 1 gives the evolution of γ exposure rate at the critical point "Ufem Horn" near this plant. In addition to the Chernobyl increase, the contribution of the Mühleberg event in september 86, has been the only case where releases from a NPS in Switzerland has induced a detectable deposition near a nuclear power station.

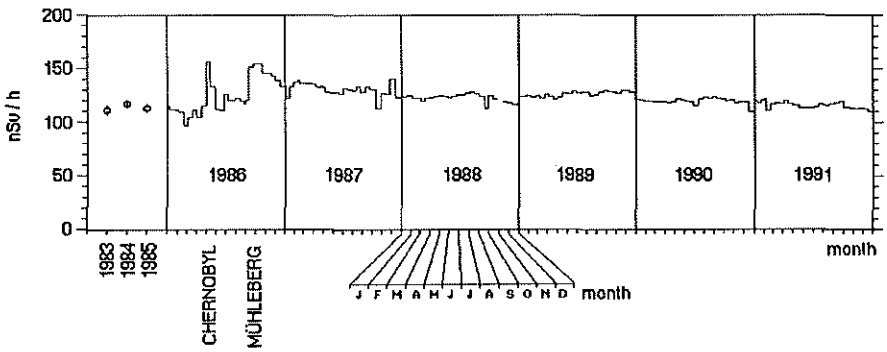

Fig. 1 Exposure rate measurements with RSS ion. chamber at the critical point "Ufem Horn" near the Mühleberg power station

2. Aerosols, vaseline sheets and rain water

Monthly evaluated values of air filters plotted in Fig. 2a show the artificial cesium-137 and natural beryllium-7 aerosol concentrations at the stations near the nuclear power plant of Leibstadt since 1984. The corresponding gross β activity of deposition on vaseline sheets (Fig. 2b) and of the precipitations shows a similar evolution. Apart from the Chernobyl contribution, these measurements indicate no influence of the releases of this plant. For the other NPS, the plots are similar excepted at Mühleberg, where nuclide specific measurements made after an incidental aerosols release in september 86 indicate traces of cobalt-60 (up to 30 μBq/m³ in air filters resp. 30 Bq/m² on vaseline sheets).

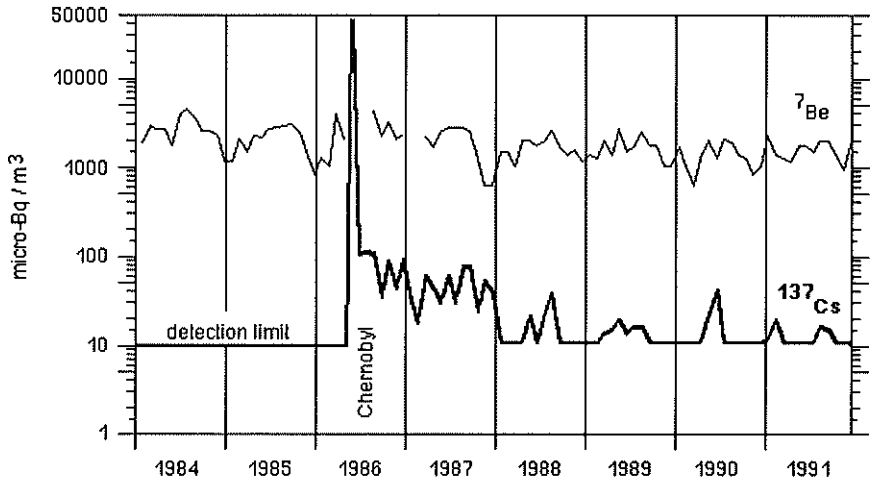

Fig. 2a ^{137}Cs and ^{7}Be concentrations in air filters at the sampling station Leibstadt.

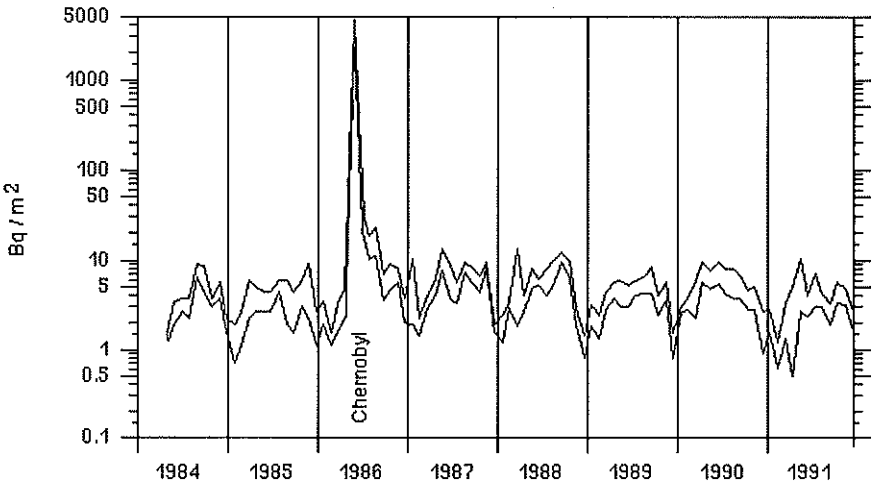

Fig. 2b Gross β deposition on vaseline sheets around Leibstadt (range of values from 4 sites).

3. Aquatic Ecosystem

Measurements performed in river (Aar, Rhin) and ground waters have not indicated an influence of releases from our NPS. In the river, gross β and tritium activity in the samples taken below the plants have not showed an increase compared with those above the plants. More interesting samples consist of aquatic plants and sediments, which accumulate long life cesium and cobalt isotopes. Samples taken downstream from nuclear power stations, where liquid radioactivity releases can contribute are therefore good indicators (Fig. 3).

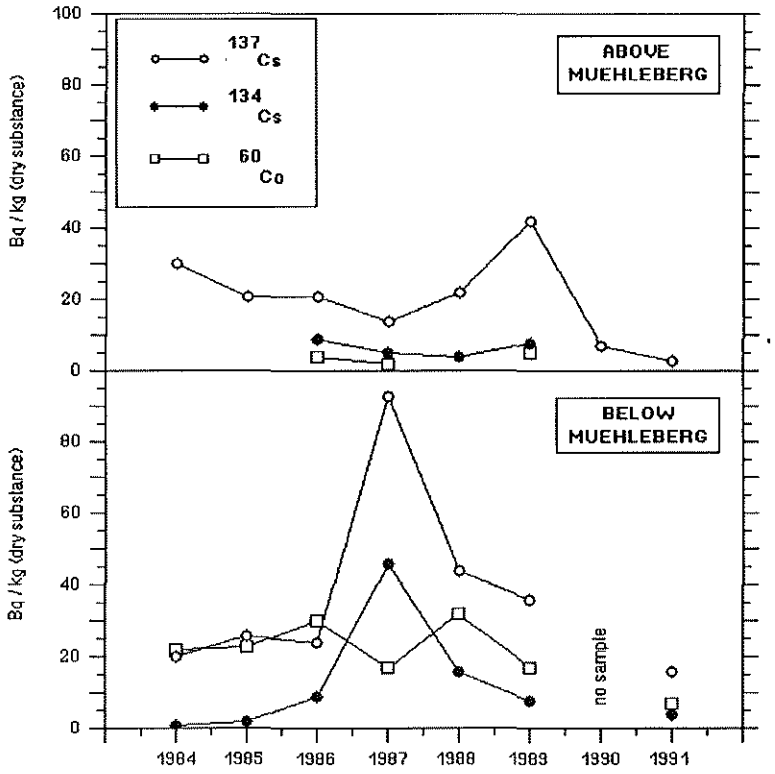

Fig. 3 Artificial concentrations in sediments taken above and below Mühleberg.

4. Soil and gras

4.1. Laboratoy measurements

Other signs of past contaminations can be obtained by nuclide specific measurements of soil samples. Apart from the Chernobyl and the atomic bomb test fallout, only the local deposition due to an unusual release of radioactivity after the Mühleberg event in september 1986 [2] has been detected in soil and grass. The interpretation of results obtained by analysis of samples in laboratory depends much on the sampling quality, the representativity the sample, depth of sampling, type of soil (cultivated or not) etc. As example, Fig. 4a shows results in soil profiles for artificial radionuclides. These measurements have confirmed a near homogeneous distribution for natural radionuclides. To get a representative mean value of a local contamination and to follow its evolution, in situ measurements are more appropiated (Fig. 4b).

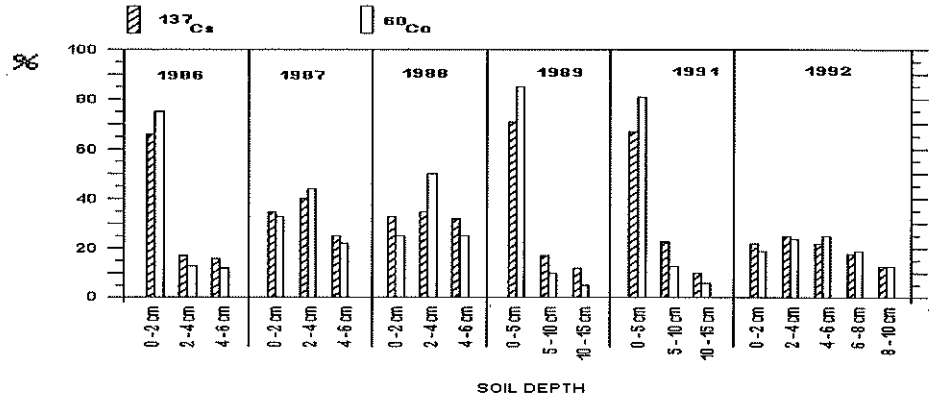

Fig. 4a Distribution of the ^{137}Cs and ^{60}Co inventory in the soil upper layers at Ufem Horn (Mühleberg)

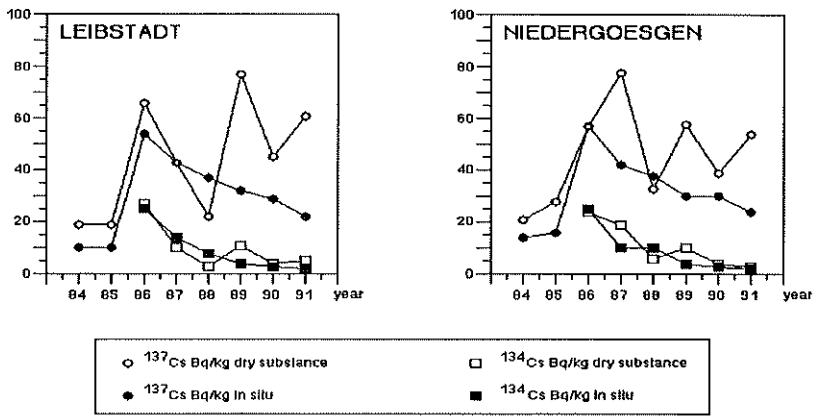

Fig. 4b Activity of the cesium isotopes at two sites near Leibstadt and Gösgen .

4.2. In situ measurements

In the last years portable Ge detectors and battery powered multichannel analyzers became available for rapid determination of the mean concentration value of natural and artificial γ emitters in the soil or on its surface. Further in situ spectrometry makes it possible to distinguish between traces from Chernobyl and atomic bomb test fallout and those from releases of NPS. This method [3] also allows to estimate the individual contribution of the identified radionuclides to γ exposure. To illustrate the trend of the exposure rate contributions, Fig. 4c shows their progress in time at one site in the immediate vicinity of each Swiss NPS in comparison with a site in the south of Switzerland (Tessin) more seriously affected by Chernobyl. One can observed a progressive diminution of the artificial component after 1986 due to the decrease of the γ emitters and to their transfer in deeper soil layers; in comparison the natural component remains nearly constant. The plotted values assume an homogeneous distribution of the radionuclides in soil permitting, from year to year, a very sensitive detection of an additional deposition. This assumption induces an overestimation of the artificial dose rate components, which is confirmed by the comparison between total γ exposure rate calculated from in situ spectra plus cosmic ray γ component and measured exposure rate obtained by ionization chambers RSS.

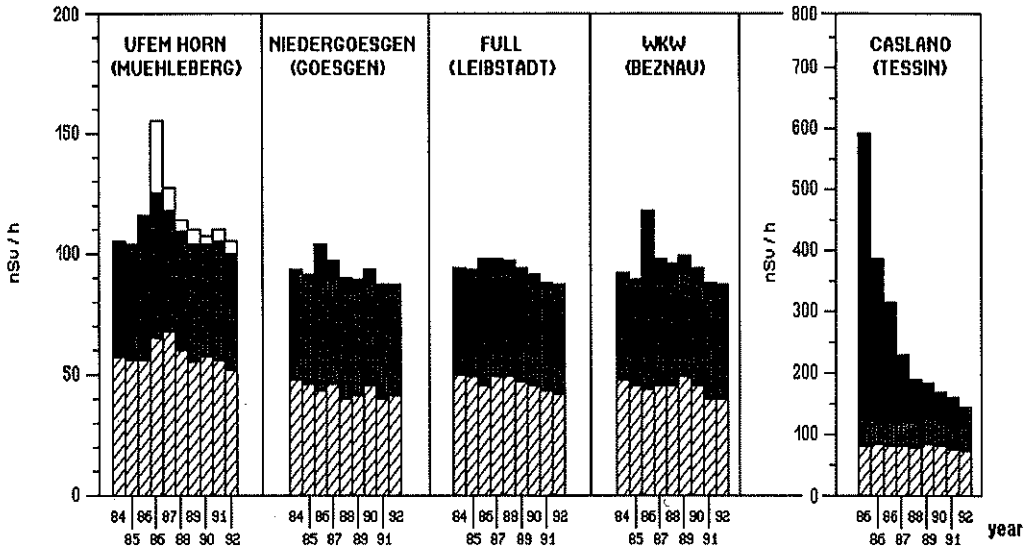

Fig. 4c Gamma exposure rate contributions 1m above ground calculated from in situ spectra, based on the conversion 1 μR/h = 10 nSv/h.

5. Other measurements (tree leaves, milk, corn and food)

Carbon-14 in in beech tree leaves, measurements performed by the University of Berne, show an increase especially near Swiss BWR up 10% of the natural concentration in comparison to the about 20% additional contribution still due to the atomic bomb test fallout. In milk, corn and other food samples, apart from the Chernobyl and the atomic bomb test fallout contributions, no additional contribution from the Swiss power stations has been registered.

6. Conclusion

The most important long lived radionuclides still remaining detectable in the environment these days are cesium-137, strontium-90, carbon-14 and tritium, since their half-lives are 30, 28, 5730 and 12 years respectively. They are mainly due to past atmospheric nuclear weapon tests (cesium-137 and strontium-90) and to Chernobyl (cesium isotopes). The radioactivity releases from Swiss nuclear power stations under normal operation give at most, for conservative assumptions, an additional dose to the surrounding population of a few percent of the natural irradiation, very small in comparison with that of other man made pollutions of the environnment. The impact of NPS in Switzerland is therefore insignificant in the environment and not detrimental to public health.

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ENVIRONMENTAL MONITORING: STATE OF THE MEASURING TECHNIQUE AND OUTLOOK

Dr. E. Frenzel
EG&G Berthold, Bad Wildbad, Germany

0. Introduction

Six years after the reactor accident at Chernobyl, monitoring the airborne radioactivity is still an important issue. Due to political pressure as well as pressure by the public, nearly all European countries have set up measuring networks for environmental monitoring.

Owing to the political changes in Eastern Europe and a better flow of information, we now have enough knowledge of the safety standards of the reactors operating in Eastern Europe and in the CIS. In addition to a discussion on upgrading the safety technology, the question of accidents and their effect on the environment is therefore still of immediate concern.

On a national and international level, monitoring nuclear facilities through large-area, interconnected measuring networks is a *conditio sine qua non* in order to take the necessary actions, to issue recommendations, prohibitions and guidelines; in a nutshell: to practice radiation protection!

1. The Radiological Variables

The purpose of environmental monitoring of airborne radioactivity is to monitor domestic and foreign nuclear facilities. Radioactivity released into the air is quickly transported to other locations. Only the fallout (wet or dry), however, will open up other paths of contamination. Thus, monitoring the air is of particularly high priority to ensure that contamination will be detected as early as possible.

Large-area measuring networks featuring a highly modern automatic measuring technique are employed [1][2]. On the on hand, the measuring systems should be as perfect as possible, and, on the on hand, they should be based on long years of successful operation. Different concepts of measuring networks have been designed to meet these requirements.

There is wide agreement, however, on which radiological variables have to be detected:

- local gamma doserate
- alpha/beta particulate
- gamma (nuclide-specific) particulate
- iodine

Together with meteorological variables (wind speed and wind direction, precipitation) which must be measured at the same time, one will then get information on airborne contamination. The dose is calculated by special computer programs from the nuclide composition and doserate measurements, and dose maps are provided. Typical for the technology employed in measuring networks is that it is a quasi fully automatic mini laboratory. The instruments are able to

- collect or accumulate
- measure
- calculate
- assess
- store
- transfer.

Of course, a radiation protection expert could do this much better, with more knowledge, more sensitive and flexible - but not quite as inexpensive, uninterrupted and automatically as an instrument operating in a measuring network! That's the compromise!

2. State of the Measuring Technique

The instrument manufacturers have developed and made instruments for measuring networks for more than 30 years [3]. Modern measuring systems of 1992 include the following features:

- microprocessors
- HPGe detectors, cooled with nitrogen or electrically
- Personal Computer
- modems for data transfer
- video monitors
- fully automatic, self-monitoring software.

Since these instruments are manufactured only in a few countries and compete on the international market, the principles of measurement and the detectors used are very similar. There are huge differences, however, in the software, the quality and price. The following instrument families are employed in nearly all measuring networks for the early detection of the essential radiological variables:

- gamma local doserate measuring stations
- particulate monitors
- iodine monitors

2a. Local Doserate Measurement

The local doserate measurement is a non-nuclide-specific method. Local doserate measurements are an excellent method of providing information covering large areas. Typical network ranges vary between 10 km and 10 km to 100 km and 100 km.

The detectors employed in these measuring networks are essentially energy-filtered, gas-filled counter tubes, mostly Geiger-Müller tubes or proportional counter tubes, occasionally scintillation detectors. A detector that is not widely used in Europe is the high pressure ionization chamber, featuring a measuring range of eight decades; due to its expensive electronics, however, it is hardly suited for environmental monitoring.

In order to cover the measuring ranges - up to 10 Sv/h - one has to use two detector systems (low and high dose range). The detectors are capable of detecting variations of a few nSv/h in a 2-hour measuring cycle with a high level of confidence. Modern detectors are moreover equipped with "local electronics" which often include the appropriate interfaces for data transfer (Fig. 1). Intelligent detectors feature data storage, record curves via remote control and perform automatic background correction.

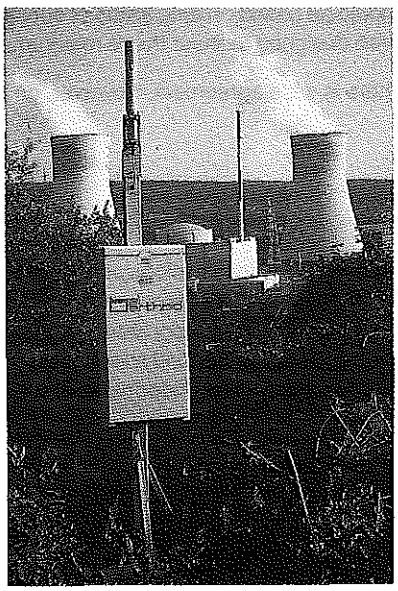

Fig. 1: γ doserate monitoring station

Due to their physical properties, proportional counter tubes are best suited for the use in local doserate measurements (low inherent background, sensitive at low energies, short deadtime). Until a few years ago Geiger-Müller counter tube were mainly used; due to the renaissance of the proportional counter tube, together with an electronics on SMD-basis, proportional counter tubes are now preferred (Telerad network, Belgium).

Measurements of the local doserate are so sensitive that even very minor variations of the natural level can be detected (e.g. after precipitation). Through combination of intelligent measuring heads with precipitation sensors one can clearly identify the smallest rise in artificial radioactivity and discriminate it from level variations caused by precipitation.

2b. α/β Particulate Measurement

An essential task of measuring networks is the early detection of airborne contamination. This method is of particular importance because this measurement is one of the fastest methods.

The measurement of airborne particulate activity concentrations is quite difficult for several reasons. The artificial activity concentrations to be detected are significantly lower than the fluctuating natural activity concentrations. On the other hand, the levels of this airborne particulate activity concentration are so low that only accumulation methods can be used.

Using powerful pumps (up to 70 m^3/h), the air is aspirated through glass fibre filter to accumulate the radioactive aerosols. Depending on the measuring task, a detector is installed above the dusting location, a large-area proportional counter tube sandwich. With counter tubes, the evaluation takes place via the AERD/ABPD technique (detection limit for artificial α's $\approx$ 50 mBq/m^3, β's $\approx$ 100 mBq/m^3) These fixed filter systems, however, permit only operation periods of max. 1 week; in other words, this method requires a great deal of servicing.

Moving filter facilities allowing automatic operation of the moving filter up to 6 months are particularly suited for the use in measuring networks. Although instruments which use the alpha-beta ratio method detect only total artificial beta (or when using the pseudocoincidence difference method, total artificial alpha as well) this method is still being used because it is the only method which permits the detection of nuclide groups which have no gamma transitions.

Modern particulate monitors with continuously moving filter feature air flow rates up to 25 m^3/h. They are capable of detecting artificial α and β activity concentrations of about 100 mBq/m^3 in several Bq/m^3 natural activity concentration level within less than 1 hour. Moving filter systems allow, moreover, to extend the measuring range up to about 10^6 Bq/m^3 by increasing the filter transport speed.

The delayed measuring system is still from those days where one could not differentiate between artificial and natural activity on-line. Its detection limit is around 500 $\mu Bq/m^3$ and it still has not lost its attractiveness in 1992, since it is able to measure fully automatically and continuously as low as possible.

2c. Gamma (Nuclide-specific) Particulate Measurement

Nuclide-specific identifications with HPGe detectors, fully automatic analysis software and, if necessary, remote data transfer facilities, are today's state of the art of science and technology. The measuring instrument of the first generation used rather complicated and problem-oriented programming languages; today's trend is clearly to use commercial languages running on standard PC's.

Nuclide-specific measuring systems, moreover, include an interface for remote data transfer and even allow the data transfer of spectra. To ensure an optimum early warning, these instruments are designed such as to allow the simultaneous alpha-beta measurement and the nuclide-specific measurement. Through integration of a PC the instruments become very user-friendly, an advantage particularly for efficiency and energy calibration. In the automatic mode, modern instruments are capable of reaching detection limits of significantly less than 50 mBq/m³ for Co-60 in 1 hour. Modern nuclide-specific monitors (Fig. 2) analyze the spectra automatically for up to 100 different nuclides. The detection limits of nuclides that were not found are calculated e.g. according to DIN 25482, Part 1 [4].

Due to the option to remote transfer all data, parameters and spectra, the diagnosis and service costs for these systems can be reduced significantly when the primary data can be analyzed in the central station. Even the "manual" evaluation of spectra or by means of software created by the user is no problem at all. Measuring system in inaccessible locations (e.g. on high mountains or islands) become independent from the nitrogen supply by using electrically cooled germanium detectors.

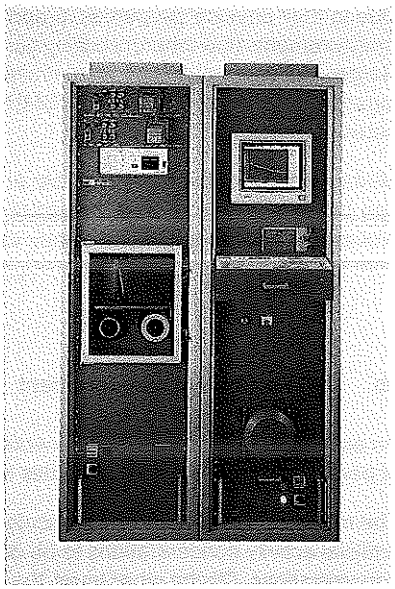

Fig. 2: Particulate Monitor for nuclide-specific measurement

2d. Iodine Measurement

Iodine (and therefore radio-iodine) is selectively accumulated in the thyroid gland. This potential health hazard requires special attention. Due to the fact that iodine and most of its compounds are very transient, special filters have to be used. Iodine occurs bound to aerosols as well as in a gaseous form. For gaseous iodine one uses special filters, such as silver-impregnated activated carbon. Unfortunately, these filters, which are particularly suited for gaseous iodine, have a low retention capability for aerosol-bound iodine.

If both iodine species are bound to one absorber, one uses special absorbers, such as AC 6120; the consequence will be, however, that filter and detector have to be heated up to prevent that the temperature will drop below the condensation point. By combining a particulate monitor with an iodine monitor for gaseous iodine one will be able to detect both species.

In the iodine monitor, the iodine is concentrated in a cartridge which includes the respective absorber. The iodine cartridge surrounds the detector either as a Merinelli container, or it is located directly next to the detector. Usually, the 364-keV peak of the parent nuclide ^{131}Iodine is detected by means of a NaI-detector. Today, one can measure ^{131}Iodine activity concentrations up to several hundred mBq/m³ per hour with high statistical confidence.

In addition to the instruments with fixed iodine cartridge, instruments with iodine cartridge changer mechanism are also being used. Due to this technique, high-resolution gamma-spectroscopy using HPGe detectors my be employed in iodine measurements as well. Instruments with iodine cartridge changer are designed such that the activation of the iodine cartridge takes place only when a certain alarm level is reached, which is detected via a standard moving filter system.

2e. Off-Line Measuring Technique

Artificial activity concentrations in the μBq/m³ range cannot be detected any more by automatic instruments; the level of the natural activity (Bq/m³) is simply too high. But in this case nature is kind enough to help us, since the aerosol-bound Radon/Thoron daughter activity is short-lived (hours), so that by "waiting" = "decaying" another chance for a measurement is provided. The price for this is "off-line", i.e. collect dust on a filter, wait and then measure.

Typical are high-volume samplers of 100 m³/h to 1000 m³. With dust collecting periods of one week and a waiting period of another week one will get detection limits of a few μBq/m³ with a 24 hour measuring time, for example, for ^{60}Co [5]. When using LUWA-Gelb as filter material, these filters can be used, after the analysis with the gamma spectrometer, for the analysis of ^{90}Strontium and ^{239}Plutonium (after ashing), as it is done, for example, by the German Weather Service.

3. European Measuring Networks

Table 1 shows a survey of the European measuring networks. The most detailed measuring networks exist in Germany, France and Holland. With regard to fully automatic data transfer France is ahead: remote transfer of data from particulate monitors, local doserate measuring stations and even spectra from nuclide-specific facilities is possible. Although huge amounts of data are created and stored due to remote transfer, the general public is in a position to get information via BTX or Minitel only in Bavaria and in some regions of France.

However, there are some steps toward a European measuring network and the comparability of data (unfortunately, only very one-sided). For example, French local doserate probes are located in some neighboring countries (Germany, Switzerland).

Eastern European countries are - still - missing! We hope that there the first steps toward measuring networks will be taken in a few years. These countries will have a certain advantage then: they may choose the best or the most economical concepts and don't have to learn it the hard way!

4. Comparability of Data

The comparability of data on a national level is not a trivial matter, but on an international level it gets to be quite complicated.

To be able to meet all international requirements, EG&G ORTEC, for example, provides eighteen different formulae for calculation of the detection limit for γ spectrosopy. Which is the correct one?

For the local γ doserate, some use the unit "μSv/h", whereas others recommend "μGy/h" instead, and we all know what our friends across the Atlantic think of SI-units. But even questions like background subtraction, correction of the data on the same altitude above sea level, subtraction of the cosmic components, are answered by each country and every user according to the best of his knowledge. Fig. 3 shows the response of various measuring systems at the same location [6]. Standardization is very much needed!

But even with particulate monitors there is a need for clarification when it comes to comparing data. In particular when setting up measuring systems and using compensation methods there is quite some confusion. The term "calibration factor" alone leaves room for many interpretations and applications. Per definition, a calibration factor is the proportionality factor including the dimension between measured value (e.g. in s^{-1}) and the physical unit (e.g. Bq/m³). In practice, these factors are determined by means of test sources (because it can't be done any other way), i.e. static. Actually, however, the calibration should be dynamic (which can rarely be done in practice)!

For example, a call for tenders may stipulate that the manufacturer, as a kind of type-test, must check and/or calibrate the instrument with artificial aerosols (e.g. ^{239}Pu and ^{137}Cs). The source tests after that are then genuine follow-up calibrations and guarantee, when done correctly, the correct reference point. This data is then comparable!

In France, there is such an institute with enough laboratory capacity and we know of similar experiments in Holland. Even the IEC has been discussing test and calibration regulations for particulate monitors for quite some time in a Working Group.

Even for the software, especially for gamma spectroscopy, there is a need for standardization. Research of the IAR Freiburg has shown that for the analysis of a given spectrum (activated natural uranium) software results by various manufacturers deviate by as much as ±30 %; however, this must be taken into account when comparing data on a national and international level.
In my view it is of equal importance for the user and the manufacturer to achieve standardization for:

- units of the local doserate measurement
- defining the results taking into account interfering variables
- defining calibration regulations, follow-up measurements
- standardization of the requirements for gamma software.

What good are highly sensitive, expensive and complicated measuring systems when everybody can interpret the results according to their own "philosophy"?

5. Conclusions

Today, low level results of some mBq/m³ are available within several hours. In addition, the measuring technique of 1992 even enables us to identify nuclides or to classify nuclide groups according to the type of radiation (alpha, beta or gamma).

Automatic measuring systems for the emission measurement of radio-iodine in the environmental air have been available for several years now. Certainly, in the future the germanium detector will replace the NaI-detector, as this is the only way to selectively detect different iodine nuclides.

In the years to come, automatic performance checks and selftests will be used on a routine basis. The increasing use of PCs makes the calibration with integrated reference sources possible.

Automatic measuring networks for environmental monitoring exist in many European countries, for example, Germany, France, Spain, Holland, Switzerland and others. The measuring network differ in details; however, nearly all networks measure the local gamma doserate, the concentration of radioactive aerosols and radio-iodine in the air. There is a general trend towards upgrading such systems to include the nuclide-specific gamma measurement.

Particularly with regard to the early detection of airborne contaminations, the measuring technique with airplanes will gain in importance. With regard to data transfer, there is an obvious trend toward wireless transmission . Some interesting options may arise as soon as the satellite communications has reached a higher technological level.

In his paper presented on the 8th Experts' Meeting on Environmental Radioactivity in Berlin in 1990, Dr. Rupprecht Maushart expressed his vision of a measuring network linked all over Europe [7]. We have come a little closer to this goal, as the common seminar of the German-Swiss *Fachverband Strahlenschutz* and the *Societé Français des Radioprotection* illustrates. The first measuring network across national borders exist already, but we still have a long way to go toward a European measuring network.

The topic of the American Health Physics Society meeting in January 1993 will be: "Environmental Health Physics". In the United States and Canada the awareness of the importance of environmental monitoring is growing rapidly. Hopefully we will discuss the vision of a worldwide linked measuring network by the end of this century.

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	γ-doserate	α/β-particulate	γ-particulate	iodine	data transm.
Austria	x	x	--	--	(x)
Belgium	(x)	(x)	(x)	(x)	(x)
CSFR	[x]	[x]	[x]	[x]	[x]
Denmark	x	--	--	--	x
Finland	x	[x]	[x]	[x]	x
France	x	x	x	x	x
Germany	x	x	x	x	x
Greece	[x]				
Italy	[x]	[x]	[x]	[x]	[x]
Luxembourg	x	[x]	--	[x]	x
Netherlands	x	x	x[1.)]	x[1.)]	x
Norway					
Spain	(x)	x	[x]	x	x
Sweden	x				x
Switzerland	x	(x)	[x]	[x]	[x]
U.K.	x	x[2.)]	[x]	x[2.)]	x

x in operation
(x) under construction
[x] under discussion
1.) 1 instrument
2.) Local Area Network

Table 1 European Networks

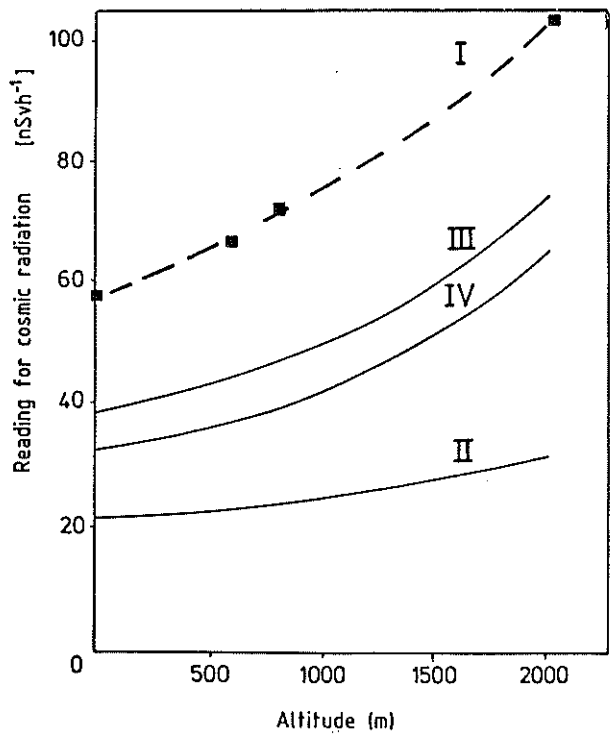

Fig. 3

Reading for cosmic radiation as a function of the altitude at the region of West Germany [6]

I Proportional Counter PD.-FHZ 600A

II Scintillation Counter

III High Pressure Argon Ionization Chamber

IV Mean cosmic radiation exposure calculated from cosmic ray flux data at geomagnetic altitude range of Germany

THE IMPACT OF NUCLEAR POWER STATIONS AND OF A FUEL REPROCESSING PLANT ON THE RHONE RIVER AND ITS PRODELTA

L. FOULQUIER, J. GARNIER-LAPLACE, A. LAMBRECHTS, S. CHARMASSON et M. PALLY.
Institut de Protection et de Sûreté Nucléaire, Département de Protection de l'Environnement et des Installations, IPSN/CEA, 13108 Saint Paul lez Durance Cédex, France.

Summary

The Rhône, with its 6 nuclear sites (17 reactors of various types and a fuel reprocessing unit), presents a relevant example for comparing the impact of these various installations on the aquatic ecosystem. Artificial radioactivity (γ emitters, Pu, ^{3}H, ^{90}Sr...) and natural radioactivity are monitored in sediments and various living organisms in the river and its prodelta. A summary of the radioecological procedure is given and illustrated with examples selected from results obtained over the last fifteen years (data resulting from about 7500 samples taken up- and downstream of the installations and in the prodelta). The evolution of results obtained during this period by γ spectrometry on fish up- and downstream of the nuclear power station at Bugey and the Marcoule fuel reprocessing unit is presented. The role of aquatic vegetation as indicator of radiocontamination is also illustrated. The evolution in the concentration levels of γ emitting artificial radionuclides in sediments and mussels in the prodelta is commented on in order to show the global radioecological impact of the Rhône in the Mediterranean sea. The analyses presented show that it is possible to quantify the influence of each source term on the total artificial radioactivity of the compartments of the ecosystem. The source terms are atmospheric fallout from early nuclear weapon tests and of the Chernobyl accident, and liquid wastes of various composition from nuclear installations.

1 Introduction

Between 1956 and 1986, a considerable number of nuclear power installations came into operation along the Rhône over a distance of 300 km. The present operations, spread over six sites, include two Fast Breeders, a Natural Uranium Gas-Graphite Reactor, twelve 900 MWe Pressurized Water Reactors, two 1300 MWe Pressurized Water Reactors and a fuel reprocessing unit. The operation of these installations causes slightly radioactive liquid and gaseous wastes to be discharged into the environment, following processing and in compliance with the legislation in force. For the last fifteen years, radioecological monitoring has assessed the impact of artificial radionuclides on the radiocontamination levels of the main compartments of the ecosystem (sediment, vegetation, invertebrates, fish). The influence of each source term can be quantified: atmospheric fallout from nuclear weapon tests (^{137}Cs, ^{90}Sr, ^{238}Pu, ^{239}Pu, ^{240}Pu), from the Chernobyl accident (^{134}Cs, ^{137}Cs, ^{131}I, ^{132}Te, ^{140}Ba, ^{103}Ru, ^{106}Ru) [1], liquid wastes from nuclear power stations (^{3}H accompanied by the eight identified radionuclides ^{134}Cs, ^{137}Cs, ^{58}Co, ^{60}Co, ^{131}I, ^{110m}Ag, ^{124}Sb, ^{54}Mn) [2], liquid wastes from the Marcoule unit (mainly ^{106}Ru) [3]. A global approach is adopted so that the information required for analyzing the impact of each installation can be obtained. This approach is based on three types of complementary information: data obtained from *in situ* studies involving a series of steps (selecting the station and the compartments to be sampled, sampling, preparing, measuring, interpreting the data), bibliographical studies and experimental studies. For some radionuclides, the results of these experiments were formalized using an explanatory and predictive mathematical model, which was validated by comparing results with the field data. Using this type of model, radionuclide transfers in a freshwater food chain can be simulated under various contamination hypotheses [4, 5]. This radioecological procedure is illustrated in this article by examples from the Bugey and Marcoule sites and the prodelta.

2 Materials and methods

2.1 Sampling stations

Around 70 sampling stations were selected between the Lake of Geneva and the prodelta in order to carry out a comparison between concentration levels up- and downstream of each nuclear site. Five monitoring stations along the Mediterranean coast provide a view of the global impact of the river. By analyzing the results obtained zone by zone, the contribution of each source term to the total artificial radioactivity of the compartments of the aquatic ecosystem can be assessed (Fig.1).

2.2 Sampling, Sample Preparation, Measuring, Data Management

Depending on the station, samples may be taken from the water, sediments, vegetation, invertebrates or fish. At inland sites, raw water samples are taken by an automatic device, sediment collected with a Berthois cone, plants gathered by hand and fish caught with an electric rod. In the sea, sediment samples are taken using wide-section core drills (625 to 700 cm^2) which provide core samples of several centimeters. Since 1984, *Mytilus sp.* samples are collected monthly [6]. The samples are prepared either immediately or after freezing to -40°C. Their preparation process (drying, incineration...) depends on the measuring method to be used (standard physical and chemical measurements, γ and α spectrometry, ^{90}Sr radiochemistry, organic ^{3}H measurement...). Using relational data base management, the entire set of site results for a precise subject are available.

3 Space-Time Interpretation of the Impact of Nuclear Installations - Some Examples

At present the data available for the Rhône and its prodelta comes from measurements carried out on 7500 samples.

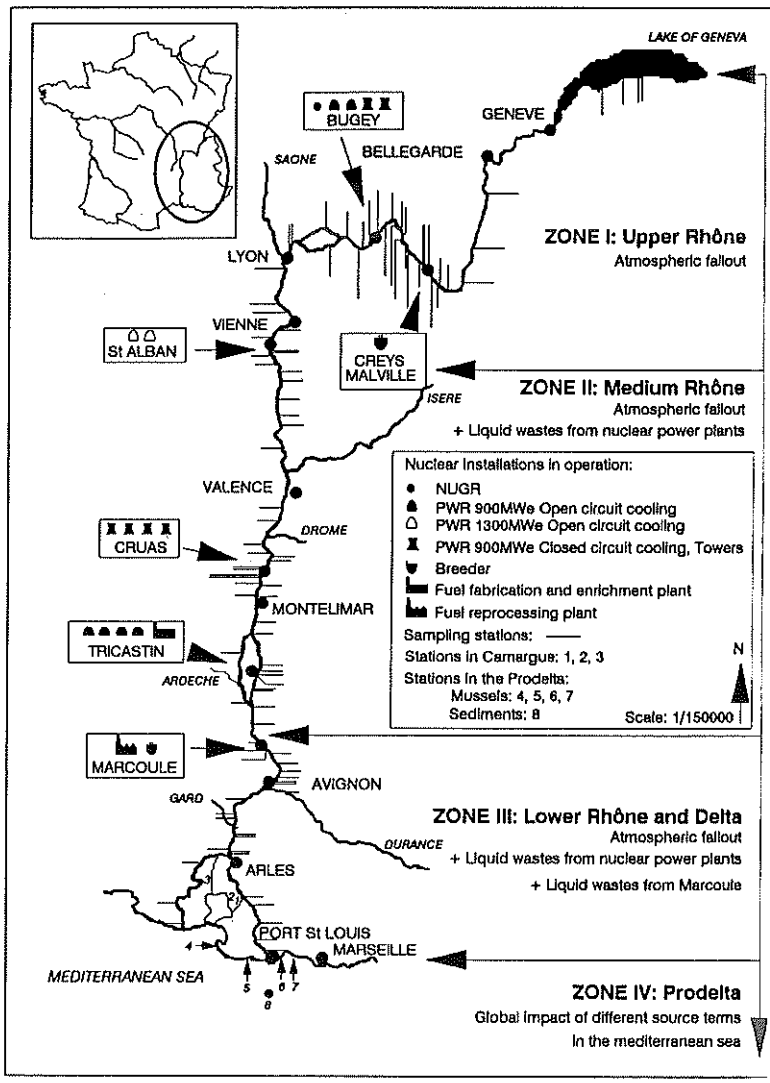

Figure 1 : The Rhône river - Location of the nuclear power stations, the fuel reprocessing unit and the main sampling stations.

By analyzing them, the evolution of natural and artificial radioactivity can be monitored over time in the various compartments of the aquatic ecosystem, and the impact of the different source terms can be compared [7,8]. The results chosen to illustrate this article were obtained by γ spectrometry on fish up- and downstream of a nuclear power station that has been operating for 18 years (Bugey) and a fuel reprocessing unit that has been operating for 30 years (Marcoule). The role of aquatic plants as radiocontamination indicators is also demonstrated for these two sites. The evolution of concentration levels of γ emitting artificial radionuclides in sediments and mussels in the prodelta is commented on in order to give a global view of the radioecological impact of the Rhône in the Mediterranean sea.

4 Illustration of the Impact of the Bugey Power Station

4.1 Evolution of Concentration Levels of γ Emitting Artificial Radionuclides in Fish (Fig.2 et 3)

From 1975 to 1985, the only source of contamination upstream of the site of Bugey was atmospheric fallout from nuclear weapon tests. This is characterized by a ^{137}Cs content of 0.22 $\pm$ 0.07 $Bq.kg^{-1}$wet

weight in fish and episodical presence of ^{95}Zr [1]. From 1975 to 1977, ^{134}Cs and ^{137}Cs were measured downstream from the site, sometimes accompanied by ^{65}Zn and ^{54}Mn. These radionuclides indicate the impact of the gas-cooled reactor. The ^{137}Cs is of double-origin: 28 % of the total concentration is due to earlier atmospheric fallout and 72 % to liquid wastes from the nuclear power station. During this period, the total concentration of γ emitting artificial radionuclides was 3.2 $\pm$ 2.1 Bq.kg^{-1}wet weight. The average obtained during the period 1978-1985 (2.4 $\pm$ 1.2 Bq.kg^{-1}wet weight), corresponding to the PWRs start (1978-1979), is not significantly different. On the other hand, ^{58}Co and ^{60}Co appear. After Chernobyl, in May 1986, a very distinct increase in cesium was recorded both up- and downstream, as well as episodic detection of ^{103}Ru and ^{110m}Ag. At that period, the impact attributable to the accident represented 95 % of the total concentration of ^{134}Cs and ^{137}Cs. From the end of 1988, artificial radioactivity levels were virtually identical to pre-accident levels (upstream: 1.2 $\pm$ 0.2 Bq.kg^{-1}wet weight in 1989). This decrease corresponds to the apparent effective half-life for radiocesiums observed in river fish, which is around 190 days [1, 9].

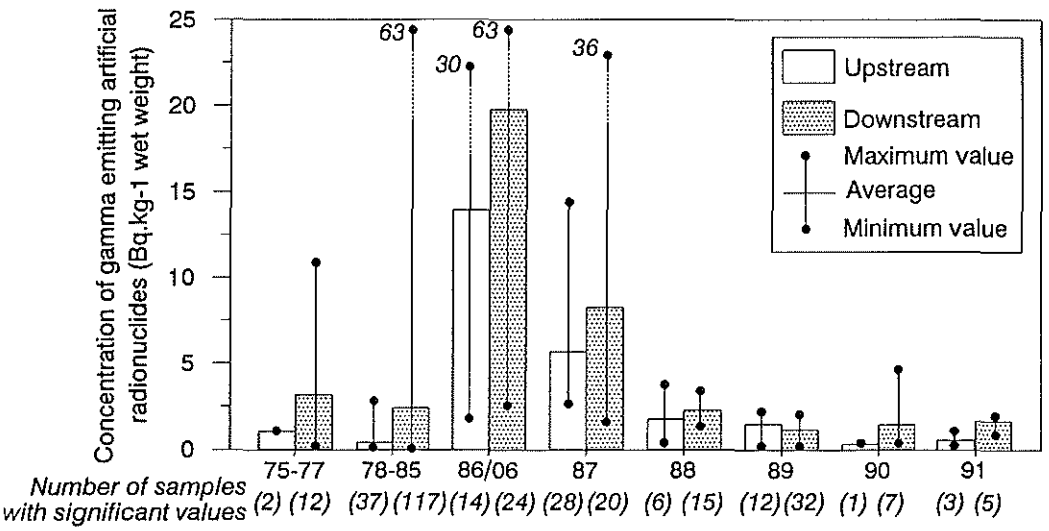

Figure 2 : Evolution with time of total concentration levels of γ emitting artificial radionuclides in fish collected up- and downstream of the Bugey nuclear power station.

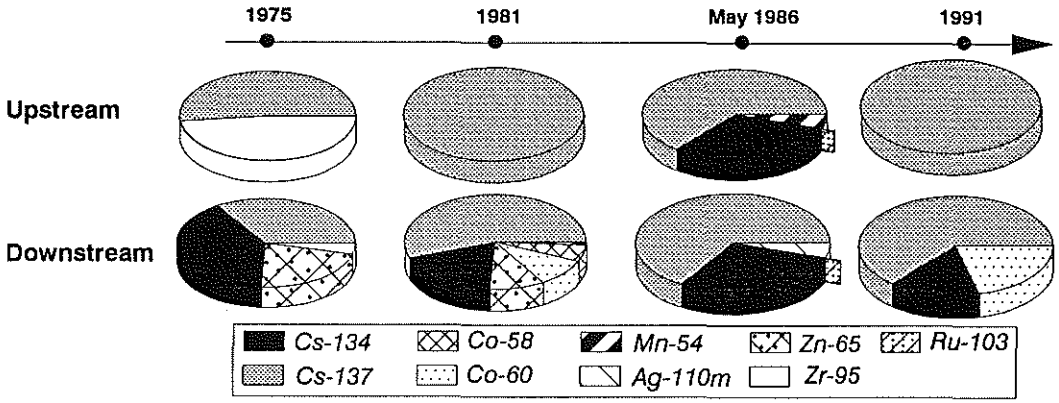

Figure 3 : Evolution with time of composition of total concentrations of γ emitting artificial radionuclides in fish collected up- and downstream of the Bugey nuclear power station.

4.2 Types of γ Emitting Radionuclides Present in Immersed Plants Collected Upstream of Bugey in 1989. Comparison with the Composition of Liquid Waste Discharges (Fig.4)

Aquatic phanerogams and mosses show a qualitatively identical spectrum of γ emitters. Compared with the annual average composition of liquid wastes, ^{124}Sb is absent because of a very small concentration factor (8 to 30 (dry weight) depending on contamination conditions). ^{131}I is absent too. The volatile

compounds of this latter separate off during sample drying but it can be measured on fresh samples which have not undergone any processing [10]. Mosses give a more representative picture of liquid waste composition than the immersed phanerogams. Generally they show concentrations which are 2 to 3 times greater (in 1989 for example, 183 to 821 Bq.kg^{-1} dry weight as opposed to 36 to 392 Bq.kg^{-1} dry weight) [11].

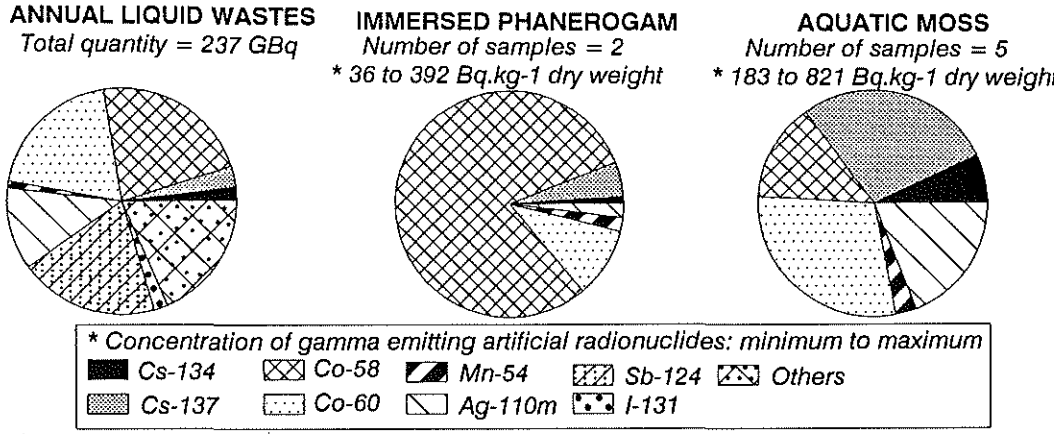

Figure 4 : Nature and composition of total concentrations of γ emitting artificial radionuclides in immersed vegetation (phanerogam and moss) sampled in 1989 downstream from Bugey nuclear power station. Comparison between the annual composition of the liquid wastes for the 8 identified radionuclides released in 1989.

4.3 Illustration of the Impact of the Marcoule Fuel Reprocessing Unit

4.3.1 Evolution of the Concentration of γ Emitting Artificial Radionuclides in Fish (Fig.5 et 6)

During the last 30 years, the operation of this unit has had a distinct effect on the various compartments of the Rhône's ecosystem. Artificial radionuclides detected upstream of the site (^{134}Cs, ^{137}Cs and episodically, ^{58}Co, ^{60}Co, ^{110m}Ag, ^{54}Mn...) appear in significantly greater concentrations downstream. Some radionuclides characteristic of liquid wastes from the unit are only detected downstream (^{103}Ru, ^{106}Ru+Rh, ^{144}Ce+Pr). As an annual average, the total concentration of γ emitting artificial radionuclides is 2 to 20 times greater downstream the site than upstream (in 1991, 0.64 ± 0.16 Bq.kg^{-1}wet weight upstream as opposed to 5.7 ± 1.6 Bq.kg^{-1}wet weight downstream). In May 1986, the Chernobyl accident produced a two-fold increase in the total concentration of γ emitting artificial radionuclides in this section of the Rhône.

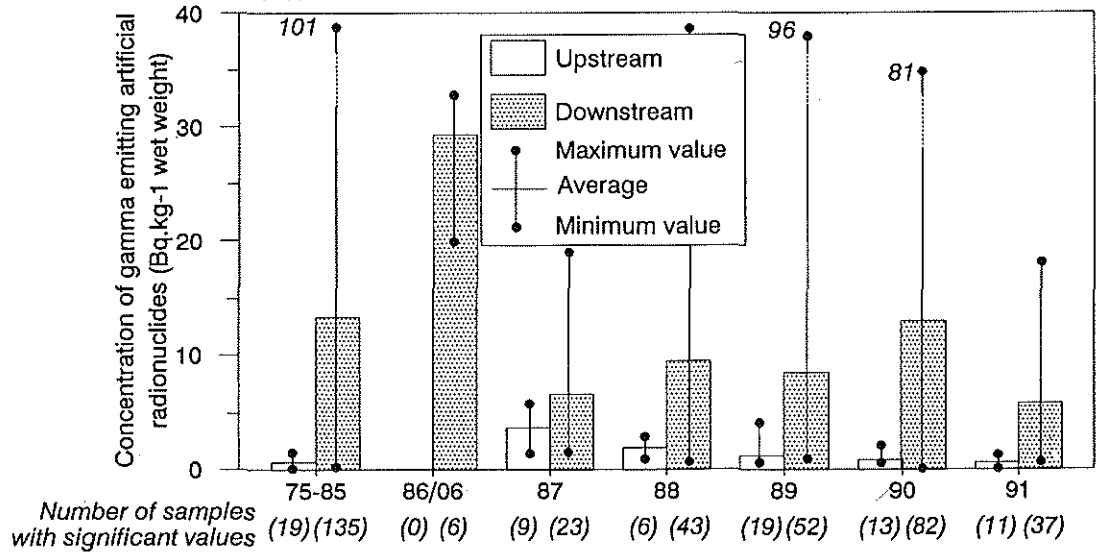

Figure 5 : Evolution with time of total concentrations of γ emitting artificial radionuclides in fish collected up- and downstream of the Marcoule fuel reprocessing unit.

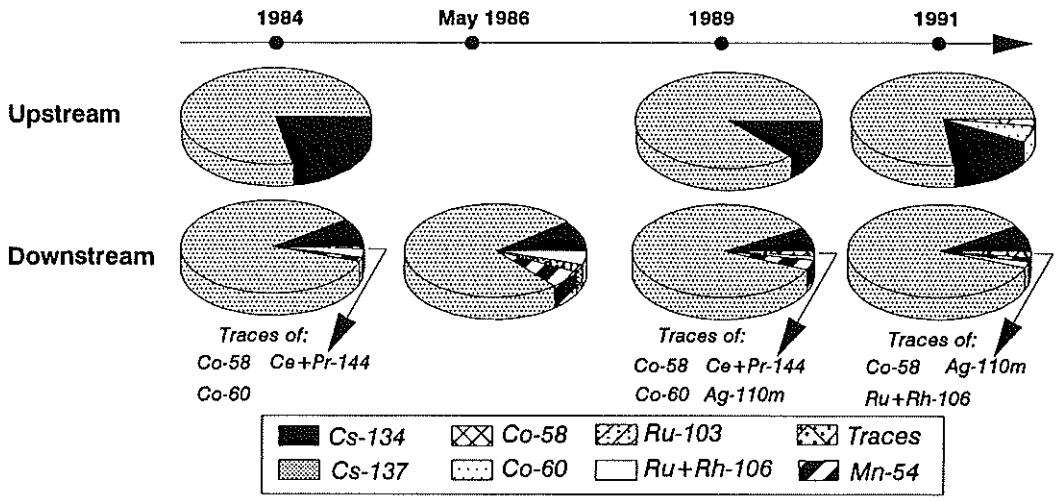

Figure 6 : Evolution with time of composition of total concentrations of γ emitting artificial radionuclides in fish collected up- and downstream the Marcoule fuel reprocessing unit.

4.3.2 Types of γ Emitting Radionuclides Present in Immersed Plants Collected Downstream from Marcoule. Comparison with the Composition of Liquid Waste Discharges (Fig.7)

The distribution of radionuclides in immersed phanerogams and mosses is very similar. Apart from ^{125}Sb, gamma emitting radionuclides present in liquid wastes from the unit such as ^{137}Cs, ^{54}Mn, $^{144}Ce+Pr$ and $^{106}Ru+Rh$ are also found in the plants. $^{106}Ru+Rh$ make up about 55 % of the total concentration in these organisms. The studies re-emphasize that the sensitivity, concentrating power and "memory" of immersed plants, and particularly mosses, make them very efficient radioactive pollution bioindicators.

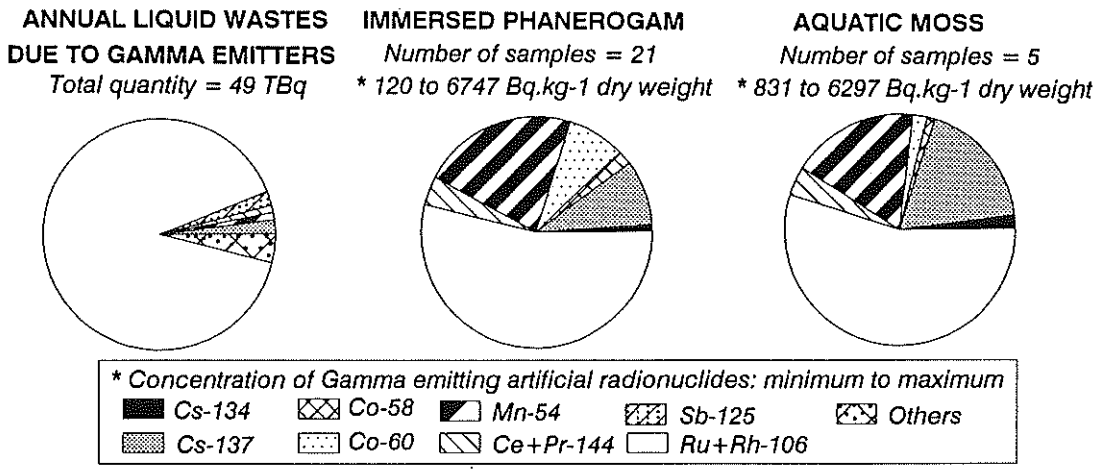

Figure 7 : Nature and composition of total concentrations of γ emitting artificial radionuclides in immersed plants (phanerogam and moss) sampled in 1989 downstream from the Marcoule fuel reprocessing unit. Comparison between the annual composition of γ emitting artificial radionuclides in liquid effluents released in 1989.

4.4 Concentrations of γ Emitting Artificial Radionuclides in the Prodelta. Examples of Sediments and Mussels (Fig.8A et 8B)

The $^{106}Ru+Rh$ observed in mussels undoubtedly originated from the Rhône before the Chernobyl period. This is confirmed by the decreasing gradient observed here and there from Faraman which is the most marked station. This element represents more than 90 % of the anthropic radioactivity due to γ emitters. The impact of Chernobyl in 1986 appears mainly as an increase of $^{106}Ru+Rh$ concentrations at all stations and systematic detection of ^{103}Ru, previously absent from the samples [12].
Several radionuclides show up in the sediments [12]. Those present in the greatest concentrations are $^{106}Ru+Rh$ and ^{137}Cs. The Chernobyl accident led to an increase in the levels of ^{134}Cs, ^{137}Cs and $^{106}Ru+Rh$. Both before and after Chernobyl, they represent 90 % of the total concentration of γ emitting artificial radionuclides in the surface sediments at this station. Following the accident, ^{103}Ru was observed in mussels but was not detected in sediments.
The prodelta data combines all waste present in the course of the Rhône. Sediments are efficient indicators of cesium discharge and to a lesser extent, ruthenium discharge, whereas mussels appear to be efficient indicators of ruthemium discharge. The absorption coefficient in sediments is twice as great for cesium as for ruthenium (2000 and 1000 $m^3.t^{-1}$ respectively), whereas the concentration factor for ruthenium in molluscs is one hundred times greater than for cesium (2000 and 20 (wet weight) respectively).

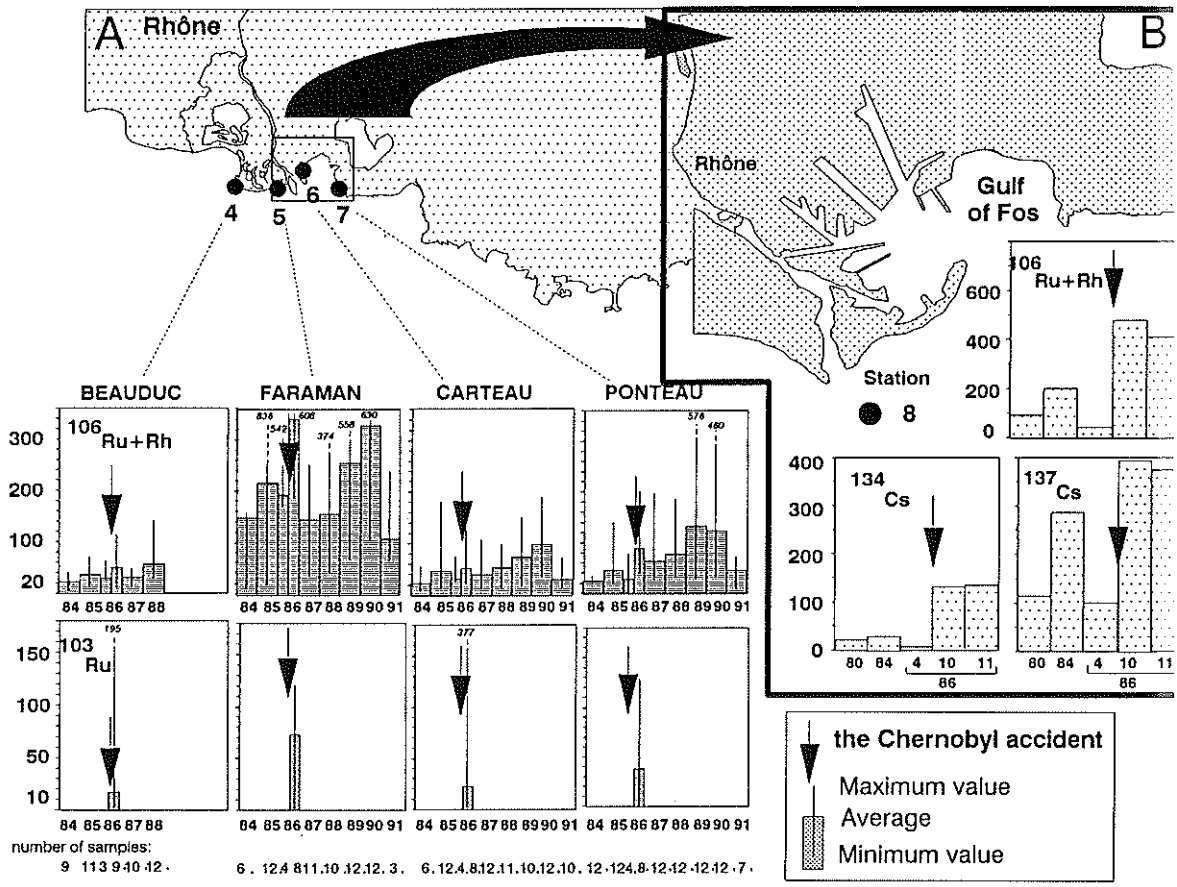

Figure 8: **A-** ^{103}Ru and ^{106}Ru concentrations (Bq.kg^{-1} dry weight: mean fresh weight/dry weight ratio around 16) in soft tissues of *Mytilus sp.* sampled at four locations in the Rhône prodelta. For 1986, the data shows concentrations recorded before and after the Tchernobyl accident arisen on 26th April.

B- ^{134}Cs, ^{137}Cs and ^{106}Ru concentrations (Bq.kg^{-1} dry weight) of sediment surface layer (0-1cm) sampled annually at various dates in front of the Grand Rhône mouth (Station 8).

5 Conclusion

Generally, artificial radioactivity due to γ emitters observed in the aquatic ecosystem downstream from a nuclear power station represents a slight proportion of radioactivity from natural sources [13]; when, since 1985, the average annual discharge of the 8 identified radionuclides from Bugey is 263 ± 54 GBq, the levels of artificial radioactivity for 1990 downstream from this site vary as follows: 0.07 to 1.6 % of the total natural radioactivity in sediments; 16 to 89 % in immersed plants (phanerogams and mosses); 0.3 to 3.9 % in fish (Tab.1). Downstream from the reprocessing unit, the percentages are higher (7.9 to 89.7 % in sediments, 6.5 to 256 % in immersed plants and 0.02 to 71.1 % in fish) (Tab.1).

Table 1: Natural radioactivity due to γ emitters in the different compartments of the River Rhône (average levels with 95 % confidence interval) and the variation range (minimum to maximum values) of total concentrations of γ emitting artificial radionuclides recorded in 1990 downstream from Bugey and Marcoule.

Compartments	Sediments	Immersed Plants		Fish
		Phanerogams	Mosses	
	Bq.kg^{-1}d.w.	Bq.kg^{-1}d.w.	Bq.kg^{-1}d.w.	Bq.kg^{-1}w.w.
Natural radioactivity	1949 ± 144 (271)*	1745 ± 291 (355)	2148 ± 541 (90)	114 ± 11 (1607)
Artificial radioactivity downstream from Bugey	1.3 to 31.2 (2)	284 (1)	1910 (1)	0.3 to 4.4 (7)
Artificial radioactivity downstream from Marcoule	153 to 1749 (9)	113 to 4642 (16)	1993 to 5572 (2)	0.02 to 81 (82)

* number of samples

Qualitatively, the main radionuclides characteristic of liquid wastes from nuclear power stations found in the aquatic ecosystem are ^{57}Co, ^{58}Co, ^{60}Co, ^{110m}Ag and ^{54}Mn. Their presence is more or less episodic and depends on the waste composition and the compartment concerned. For Marcoule, the γ emitters are mainly ^{106}Ru+Rh, ^{103}Ru and ^{144}Ce+Pr. Radiocesiums have several sources (atmospheric fallout and nuclear installations) and are detected almost systematically.
This article only takes into account γ emitting artificial radionuclides. A similar analysis could be carried out for other artificial radionuclides such as ^{90}Sr or ^{238}Pu, ^{239}Pu, ^{240}Pu, present at trace levels along the entire length of the Rhône (atmospheric fallout from earlier nuclear weapon tests), and especially downstream from Marcoule (liquid wastes from the unit) [14, 15]. Bound organic tritium also shows up in all compartments of the aquatic ecosystem downstream from nuclear installations, although it has various sources [16].
Radioecological monitoring of a river with several nuclear sites and of its prodelta provides valuable feedback of experience for evaluating the impact of liquid wastes from nuclear installations on the aquatic ecosystem. Interpretation of site results raises questions concerning radionuclide transfer mechanisms in the various ecological compartments - questions which will be answered by designing appropriate laboratory experiments. Finally, mathematical models based on experimental results and validated by *in situ* data, can be used to simulate transfer in a defined ecosystem and complete the set of tools required for analyzing a given radioecological situation.

6 Références

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IMPACT OF INDUSTRIAL NUCLEAR RELEASES INTO THE ENGLISH CHANNEL

P. GERMAIN and P. GUEGUENIAT

Institut de Protection et de Sureté Nucléaire
Département de Protection de l'Environnement et des Installations
IPSN/CEA, 50444 BEAUMONT-HAGUE CEDEX - FRANCE

Abstract

Indicator species, seawater samples and sediments are used to study the distribution and transfer mechanisms of artificial radionuclides in Channel waters. These radionuclides come from nuclear power stations along the coast, and from the very low level releases of the nuclear fuel reprocessing plant at La Hague. The observed pattern of radiolabelled zones is in good agreement with a hydrodynamic model for the Channel. The variations of activity with time are discussed in relation to releases from La Hague.

Introduction

Artificial radionuclides entering the English Channel and southern North Sea (near the Straits of Dover) are mainly derived from releases associated with the french electronuclear industry (power stations at Flamanville, Paluel, Penly and Gravelines), the nuclear fuel reprocessing plant at La Hague as well as various installations in the United Kingdom (Fig. 1). Releases from the Sellafield plant do not appear to have a detectable effect on Channel waters[1], while the spike from the Chernobyl accident is only of weak intensity[1,2]. The monitored release of weakly radioactive wastes from nuclear installations is strictly supervised in France by the SCPRI as well as the respective specialized departments of COGEMA, EDF and the French Navy. These bodies have the task of monitoring levels of radioactivity in the vicinity of each point of release. The Laboratoire d'Etudes Radioécologiques de la Façade Atlantique (LERFA - La Hague) has the objective of ascertaining the fate of released radionuclides whatever their level of activity. This means that the work of LERFA extends over a wide geographical area (i.e. the entire Channel and North Sea). LERFA is also involved in defining the various radiolabelled zones within these sea-areas and accounting for the spatio-temporal distribution of radiotracers in the different physical and/or biological compartments.

Monitoring along the french coast of the Channel shows ^{125}Sb, ^{106}Ru-Rh, transuranics, ^{99}Tc, typical of La Hague releases. At some places, near coastal nuclear power stations, the concentration in bioindicator species of some radionuclides shows higher values (such as ^{60}Co) or is significant (e.g. $^{110m}Ag < 50$, $^{54}Mn < 5$, $^{58}Co < 10$ Bq kg^{-1} dry weight)[3].

The present investigation describes the impact of industrial releases on the levels of radioactivity measured in Channel waters ; this work is based on numerous studies carried out by LERFA. Several illustrative examples are given here, but a report will shortly appear giving a more complete set of results.

Spatial distribution of radionuclides

Particular attention has been paid to the measurement of radioactive tracers showing a conservative behaviour in seawater ; this is because such tracers (^{125}Sb, ^{137}Cs, ^{99}Tc, etc.) make it possible to study the displacement of water masses. The most complete mapping of ^{125}Sb distribution yet obtained in the Channel is based on observations carried out during June 1986[1] (Fig. 1). From this survey, it can be seen that radionuclides released from the La Hague plant are mainly carried off towards the east. A pattern of parallel bands is apparent, with a strong activity decreasing gradient observed between the English and French territorial waters. These observations are supported by measurements of gamma-emitters and ^{99}Tc in bioindicator species[2,3,4]. The spatial distribution of

activities is the result of transport characteristics and dilution phenomena, as indicated by the ^{106}Ru-Rh distribution in *fucus* (5) compared with the pattern of residual lagrangian circulation and the long-term flow trajectories derived from mathematical modelling (Fig. 2)(5). About 95% of the releases from La Hague are dispersed towards the east, whereas 5% are transported westwards (1)(6).

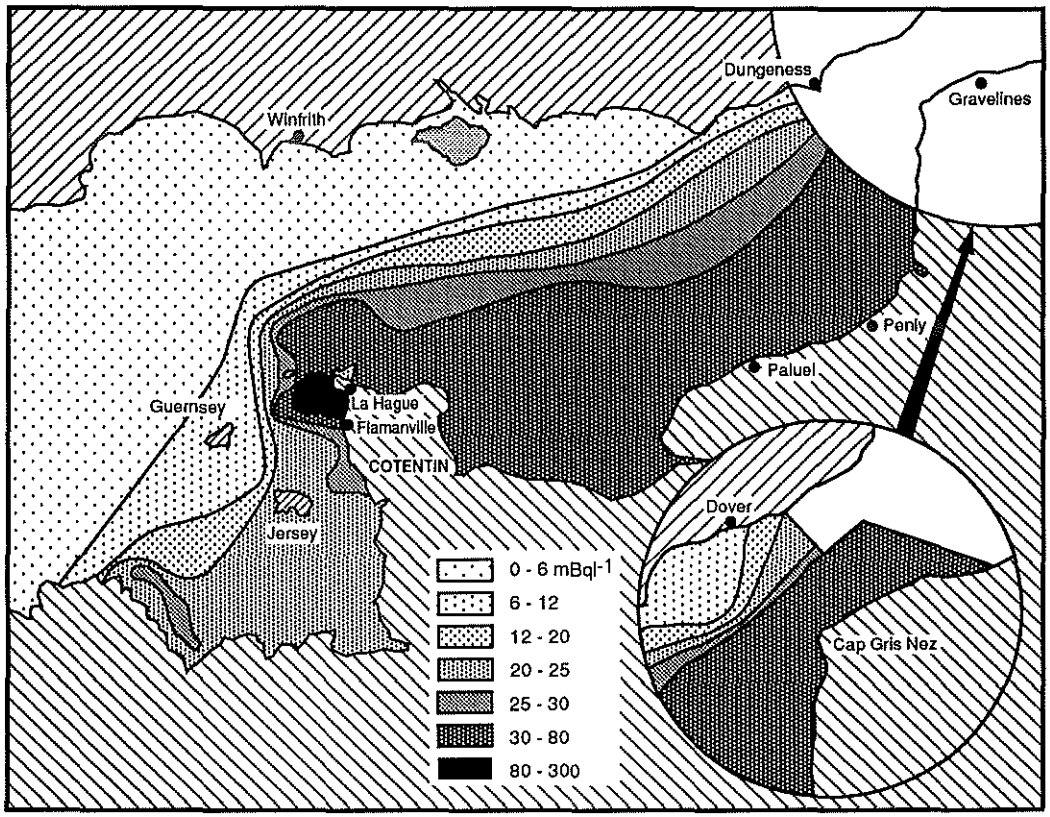

Figure 1. Distribution of ^{125}Sb activity in the central and eastern Channel during June 1986 (1)

As regards the distribution of gamma-emitters in the bioindicators, some particular features are observed which reveal the existence of zones where activity does not fall off as a function of distance from the point source (e.g. western coast of the Cotentin, western baie de Seine and coast off the Pays de Caux)(3). However, it would appear that these "radiolabelled zones" are not always so clearly defined for soluble radionuclides such as ^{99}Tc and ^{90}Sr(3), particularly along the western coast of the Cotentin. On the basis of sediment analyses, Guegueniat *et al.* (7) have shown that the radionuclides have different distributions according to their physico-chemical behaviours ; most of the soluble elements move eastward from La Hague, whereas the particulate elements are mainly deposited in the west.

In addition to the parameters described above, the spatial distribution in bioindicator is dependant on factors which are inherent to the species themselves (i.e. ecology, ethology, transfer kinetics, etc.). In limpet (*Patella sp.*) flesh samples from Dielette (15 km from La Hague), the ^{106}Ru-Rh activity fell from 51 to 14 Bq kg^{-1} dry weight (mean value X = 26) during the period from January to December 1991. Similarly, samples from Wimereux (280 km from La Hague) show a decrease in activity from 53 to 12 (X = 27) Bq kg^{-1} dry weight over the same period ; for $^{239+240}$Pu, the annual mean activity in 1991 was 0.14 Bq kg^{-1} at Dielette and 0.20 Bq kg^{-1} at Wimereux. On the other hand, the *Fucus serratus*

samples from Dielette show systematically higher activities than at Wimereux (annual mean activity of ^{106}Ru–Rh in Bq kg^{-1} is 43 at Dielette and 22 at Wimereux (Table 1) ; the $^{239+240}$Pu activity is 0.46 at Dielette and 0.18 at Wimereux, in 1991).

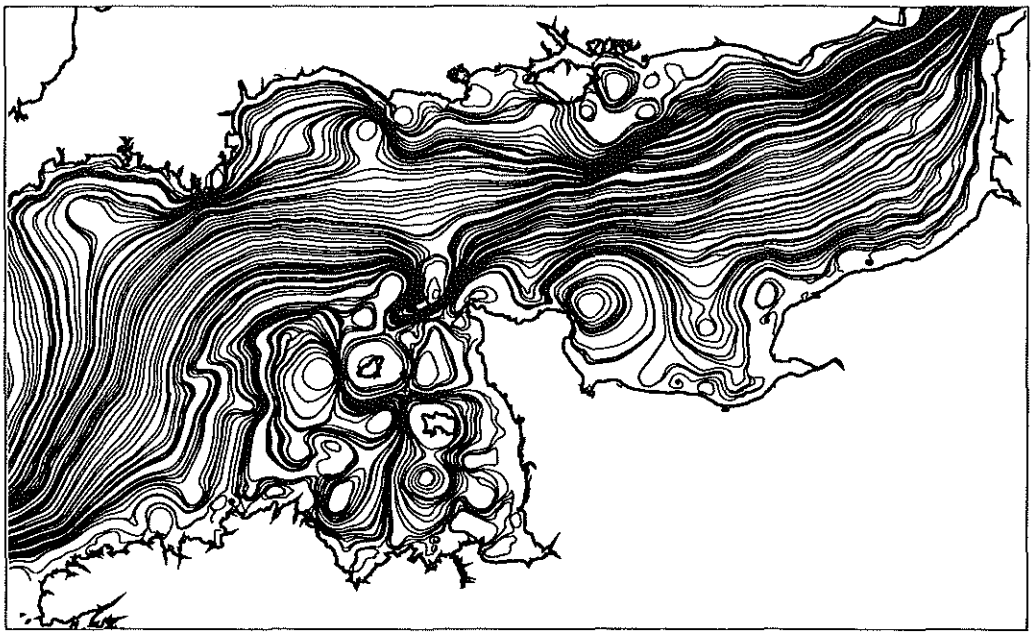

Figure 2 – Pattern of water masses identified from flow trajectories (WSW winds) (5).

Table 1 – Comparison of mean annual activities of ^{106}Ru–Rh, ^{60}Co, ^{125}Sb and ^{99}Tc for 1987 and 1991 in releases from La Hague, in seawater samples in *Patella sp.* (flesh) and in *Fucus serratus* from the north Cotentin coast and Wimereux.

Radio nuclide	Year	Annual release (TBq)	Seawater activity North Cotentin (mBq l^{-1}) (1)	*F. serratus* (Bq kg^{-1} dry weight) North Cotentin (2)	*F. serratus* (Bq kg^{-1} dry weight) Wimereux	*Patella sp.* North Cotentin (Bq kg^{-1} dry weight)
^{106}Ru–Rh	1987	1053	505 ± 132	350 ± 97	33.0 ± 15.0	293 ± 79 (3)
	1991	35.7	65.6 ± 38.5	42.8 ± 29.7	21.7 ± 4.1 (* a1)	25.8 ± 11.8 (2) (* a2)
^{60}Co	1987	7.5	6.3 ± 6.9	112 ± 27	13.9 ± 4.9	19.1 ± 5.2 (3)
	1991	2.7	8.2 ± 8.7	48.9 ± 16.5	9.7 ± 2.6	11.8 ± 5.3 (2)
^{125}Sb	1987	183	128 ± 39	7.8 ± 3.3	2.4 ± 1.1	4.6 ± 1.1 (3) (* a3)
	1991	25.1	28.0 ± 11.9	2.2 ± 0.8	1.1 ± 0.2 (* a4)	ND (2)
^{99}Tc	1987	15.9	9.0 ± 4.8	1723 ± 464	NA	NA
	1991	0.9	2.4 ± 0.9	365 ± 147		

ND : not detected ; NA : not analysed ; 1 : Goury ; 2 : Dielette ; 3 : Herquemoulin. * Mean calculated from measured values (M = number of analyses above detection limit) a1 : M = 4, ND = 8 ; a2 : M = 10, ND = 3 ; a3 : M = 6, ND = 2 ; a4 : M = 3, ND = 9.

Activity levels in English Channel waters and variations with time

Very recently, the processing of analytical results from seawater samples taken near the La Hague waste outlet has shown that the influence of radionuclide releases on the level of activity in these waters takes place in two stages : a short-term effect is observed about two weeks after emission into the marine environment, whereas another distinct phase can be recognized several months after the initial release. These two stages are controlled by the displacement of regional water masses[8]. The effect measured in bioindicator species is also observed about a few months after the initial release[9]. Although many ecological and physiological parameters can influence the level of radioactivity[3,10,11], fluctuations in releases from nuclear installations constitute the main factor controlling the variations of activity with time. The decrease in quantities of radionuclides released from the La Hague plant is conditioning the current trend towards a lowering of radionuclide concentrations in various compartments of the natural environment.
In the light of the above, a clear drop in levels of ^{125}Sb, ^{106}Ru–Rh and ^{99}Tc has been noticed in seawater off the northern Cotentin since 1990 (Table 1). The activity of plutonium isotopes is estimated at only a few μBq l^{-1} during 1992. ^{237}Np levels of the order of 10–30 μBq l^{-1} [11] and tritium levels of the order of 3.5 Bq l^{-1} are recorded in this area. ^{137}Cs activities are higher than those characteristic of fallout material (> 5.6 mBq l^{-1} and < 15 mBq l^{-1}). ^{40}K activities (natural element) are about 12000 mBq l^{-1}.
Recent results from bioindicators (Table 1) also indicate a clear decrease in levels of ^{106}Ru–Rh and ^{99}Tc, coupled with a more limited decrease in ^{60}Co and ^{239} + ^{240}Pu. During early 1992, radionuclide activities in *F. serratus* collected on the northern Cotentin coast were as follows (units are Bq kg^{-1} dry weight) :

– ^{106}Ru–Rh : < 40 ; ^{60}Co < 60 ; ^{137}Cs < 3 ; ^{99}Tc < 600 ; ^{239} + ^{240}Pu < 1 ; ^{125}Sb < 5 ; ^{40}K : 1,000–1,500.

Levels of ^{129}I are lower than 10 Bq kg^{-1} dry weight (1984)[12], while ^{90}Sr activities are less than 8 Bq kg^{-1} dry weight (1988). In all bioindicator species, levels of ^{237}Np are beneath 200 mBq kg^{-1} dry weight (1986)[11]. As regards ^{60}Co, different trends can be seen between seawater and indicator species ; this is reflected in the higher seawater activities observed in January and February, giving values which put a weighting on the annual average.

The dominant radionuclide present in the sediments is ^{106}Ru–Rh ; ^{144}Ce–Pr also used to yield some high activities, but it ceased to be analysed towards the end of the 1970s. Over the past few years, levels of ^{106}Ru–Rh have dropped off sharply, with values in the northern Cotentin falling from 1,500–3,000 to a few tens of Bq kg^{-1} dry weight.

The trend towards lower activity levels can be profitably used to gain an understanding of the kinetics of radionuclide transport in the Channel as well as the elimination of these elements in sediments and living organisms. This will enable the development of models for elemental transfer in Channel waters. This type of modelling should also involve the incorporation of hydrodynamic, physiological and ecological effects, as suggested by the differences existing between the ratio of the activity of wastes released in 1987 and 1991 compared with those of the levels recorded in various components of the environment at the same periods (Table 1).

An extensive set of different pathways – including aerosols[13] – were taken into consideration to describe the transfer of radionuclides towards humans. The radiological impact is very low[14], being less than 1% of the admissible dose. Neither the processing of seaweed during the extraction of gelling compounds nor food preparation techniques are capable of enriching the activity of radionuclides in consumer products[15].

Footnote

The recent analytical results mentioned in this article were obtained by staff at LERFA (P. Guegueniat, P. Germain, M. Masson, D. Boust, A. Fraizier, P. Bailly du Bois, R. Gandon, R. Léon and G. Leclerc).

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STUDY OF THE LONG-RANGE EFFECTS OF RADIOACTIVE EFFLUENTS FROM NUCLEAR POWER PLANTS IN THE CASE OF THE RHINE RIVER

H. Mundschenk and W. J. Krause
Federal Institute of Hydrology, Kaiserin-Augusta-Anlagen 15-17, POB 309, D-5400 Koblenz, Germany

Summary

Radioactive substances released into the rivers with the wastewater by nuclear facilities during normal operation lead to narrowly limited contamination clouds which are still detectable over long distances from the location of the emitter. Therefore, radioactive discharges can be used like tracers to describe their propagation behaviour within the aquatic reach. Thus, intermittent discharges of ^{3}HHO can be used to determine flow times and values characterizing the longitudinal dispersion of soluble radioactive substances under natural conditions. By means of the two cobalt isotopes ^{58}Co and ^{60}Co, the particulate nuclide transport in flowing waters can be registered and their propagation and sedimentation behaviours can be described.

1. Introduction

At present, in the Rhine river basin several nuclear power plants (NPPs) and other nuclear facilities are in operation in France, Germany and Switzerland. The large area surveillance of the Rhine river and its tributaries, that is realized by the Federal Institute of Hydrology according to the German Radiation Protection Contingency Act and by the competent measuring stations of the Länder according to the Rule for Emission and Immission Surveillance of Nuclear Facilities, is necessary in order to register the effects within the water bodies under normal conditions as well as in the case of increased releases [1].

In normal operation, nuclear facilities discharge radioactive substances in soluble and particulate species with their wastewater into the receiving stream. There they spread over the compartiments water, suspended matter and sediment and are occasionally detectable even at large distances from the emitter. The main components of the nuclide mixture, ^{3}H (tritium), ^{58}Co and ^{60}Co, are transported as soluble (^{3}HHO) or as predominantly particulate species. Highly sensitive detection equipment (LSC after electrolytic enrichment and high resolution gamma-spectrometry respectively), allows to use these nuclides under defined conditions as tracers in the investigation of hydrologic processes.

2. Long-range effects of soluble radionuclides

Radioactive wastewaters from nuclear power plants (NPP) are usually released over periods of 1 - 2 hours at irregular intervals. These discharges produce a labelling of flowing water bodies. The resulting increases in concentration occur in narrowly limited swells (contamination clouds) which travel along the river and can be registered at measuring and sampling points downstream of the emitting location in form of so-called transition curves. Since 1985 detailed investigations have been carried out on the rivers Main and Rhine using tritium, on the one hand, as artificially introduced tracer and, on the other hand, as labelling nuclide from nuclear power plant discharges in order to determine flow velocities, flow times, and the longitudinal mixing of the water under natural conditions [2, 3].

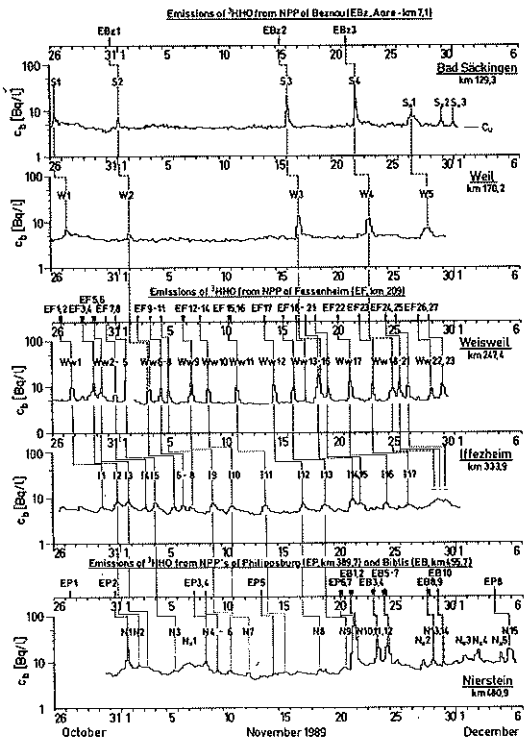

Fig.1 Tritium concentration c_O in the Upper Rhine in October/December 1989 at the sampling stations of Bad Säckingen, Weil, Weisweil, Iffezheim and Nierstein. Sampling interval: 2h/sample (Bad Säckingen) and 4h/sample (other stations)

During an investigation carried out in October/November 1989 on the Upper Rhine, ^{3}HHO discharges from the Beznau NPP could be detected in 2h-mixed samples at the Weil sampling site approximately 75 km downstream and in particular cases even after 240 km at the Iffezheim sampling site (km 333.9) (Fig.1). At the Weisweil (km 274.4) and Iffezheim sampling stations additional transition curves could be observed originating from effluents of the Fessenheim NPP (KKF) in France. Some 260 km downriver (Nierstein sampling point) they were often superimposed by effluents from other nuclear power plants located along the river. Effluents from the Philippsburg and Biblis NPPs could be registered sensitively (above 0.5 TBq) at Nierstein (91 km and 25 km downstream of the emitters, respectively). Similar investigations have been continued in three measuring campaigns in 1990, 1991, and 1992 on the free-flowing Rhine section downriver of Iffezheim to the Lower Rhine. These measurements are evaluated at present.

For routine measurements of the tritium concentration monthly mixed samples were taken as a basis for balance computations. Because of the high number of emitters and due to the relatively high river discharge, the impacts of single nuclear power plants onto the Rhine are no more detectable, whereas this is still possible on the Moselle and Main rivers.

The long-term development of the tritium load [4] can be taken from a plot of discharge-weighted annual means of tritium concentrations at several sampling points along the flow course between the Alpine Rhine (Schmitter) and the Lower Rhine (Emmerich and Bockum, since 1980) (Fig.2):

- Proceeding from the background load in the Alpine Rhine, which is a result of the so-called bomb tritium originating from the atmospheric nuclear weapon tests performed until 1963, the tritium concentration increases along the flow course due to the effluents of riparian emitters. The highest values are regularly measured downstream the mouth of the Main river in the Middle Rhine. Excepting the Moselle, the inflow of

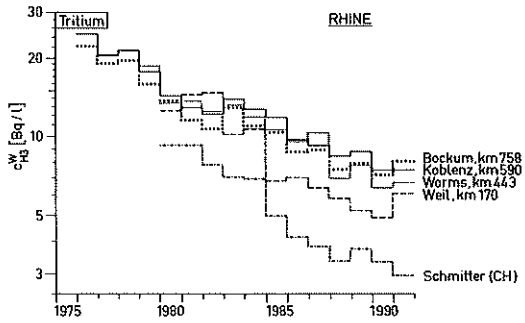

Fig.2 Annual flow-weighted means of tritium concentration c_{H3} in the Rhine at several sampling stations for the interval 1976 - 1991

tributaries on the Lower Rhine with lower tritium contents leads to a slight decrease in the tritium concentration, due to dilution.

- In spite of the intensification of nuclear power generation in the Rhine river basin over the last 17 years, a decrease of the tritium concentration in the Rhine could be observed generally. This phenomenon is based on the physical decay and on the progressive dilution of this nuclide within the hydrological cycle through less tritium containing precipitation.

- The annual means of tritium concentrations found at present in the Rhine and its tributaries fall below the limit of detection of 10 Bq/l set in the various monitoring programmes, sometimes considerably.

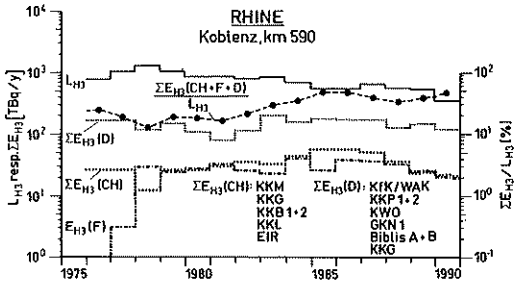

Fig.3 Annual tritium loads L_{H3} in the Rhine at Koblenz and emission rates for Swiss ($\Sigma E_{H3}(CH)$) and German ($\Sigma E_{H3}(D)$) nuclear power stations and the French NPP at Fessenheim ($E_{H3}(F)$)

In Fig.3 the shares of the annual tritium discharges from German ($\Sigma E_{H3}(D)$) [5] and Swiss ($\Sigma E_{H3}(CH)$) [6,7] nuclear power plants and the French Fessenheim NPP (KKF) ($E_{H3}(F)$) [8] are compared with the annual tritium loads L_{H3} of the Rhine at the Koblenz sampling station, upstream the Moselle inflow. Although the discharges from Swiss nuclear power plants show a slightly rising tendency, and those of German installations were stagnating, the tritium load of the Rhine has generally been regressive for 13 years now. So, the tritium load of 1260 TBq/y calculated for 1978 at Koblenz decreased to 340 TBq/y in 1990. In the same period the share of the tritium load originating from nuclear power plants was rising from 13 % to approximately 45 %. In 1990 the German share herein amounted to 74 % while the Swiss contribution and that of the Fessenheim NPP made up nearly 13 % each.

The tritium loading of the Rhine by the different emitters along the flow course is shown in a plot where the annual loads are depicted exemplarily for 1990 at several sampling points (Fig.4). For comparison, the emission rates reported by the individual nuclear power plants [5,9] for this year are also shown. Due to the fact that the emission rates of the French Cattenom NPP were not available, corresponding emission rates were estimated from existing tritium concentration and discharge values of the Moselle river at Koblenz before and after Cattenom NPP started operation. The tritium discharges of the Karlsruhe Nuclear Research Centre, the highest ones until 1989, were for the first time exceeded by those of the Cattenom location.

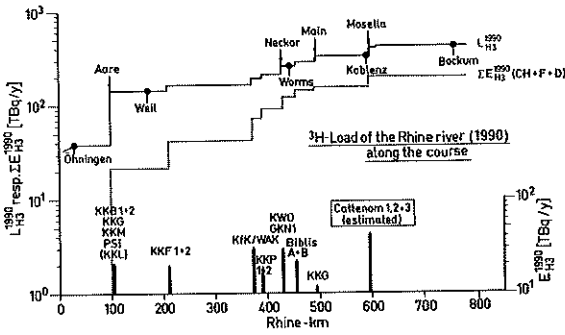

Fig.4 Tritium load L_{H3} in the Rhine river in 1990 determinated at Öhningen, Weil, Worms, Koblenz and Bockum in comparison with the annual emissions E_{H3} of Swiss, French and German nuclear facilities

3. Long-range effects of radionuclides transported in particulate form

The predominant number of radionuclides released from nuclear facilities into the receiving waters have a high affinity to sorption on suspended matter transported within the flowing water. This is especially valid for the activation products ^{58}Co and ^{60}Co, present in the wastewater of some nuclear installations as main components besides tritium. They accumulate onto suspended matter at more than 90 %, are transported together over large distances, and lead to the contamination of the sediment [10,11], preferently in high-sedimentation areas (groynes, weirs, harbours).

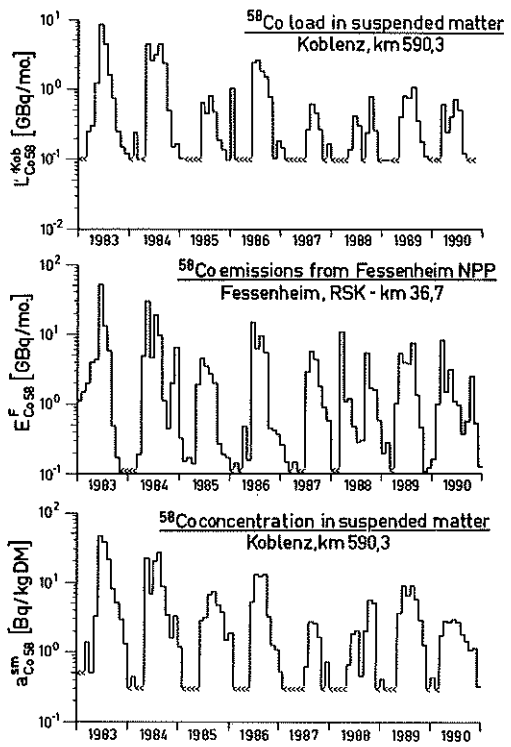

Fig.5 Concentration a^{sm} of ^{58}Co in suspended matter and ^{58}Co-load L^{Kob} in the Rhine river at the Koblenz sampling station in comparison with the emissions E_{Co58} of the Fessenheim NPP for the interval 1983 - 1990

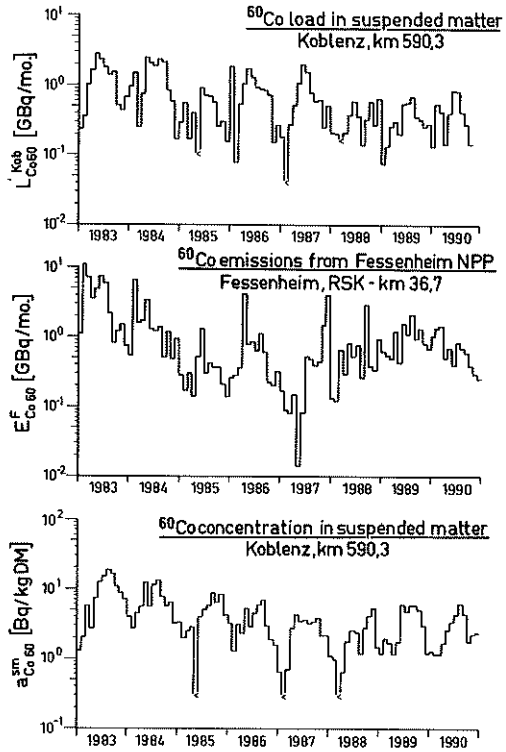

Fig.6 Concentration a^{sm} of ^{60}Co in suspended matter and ^{60}Co-load L^{Kob} in the Rhine river at the Koblenz sampling station in comparison with the emissions E_{Co60} of the Fessenheim NPP for the interval 1983 - 1990

Both nuclides are measured along the Rhine course in concentrations underlying regular seasonal variations. The ^{58}Co and ^{60}Co concentrations of monthly mixed samples are shown for the Koblenz sampling site (Figs.5 and 6), where even after a distance of approximately 380 km and a discharge-depending transport time of 5 - 10 days, the ^{58}Co content of the suspended phase reflects the monthly inputs from the Fessenheim NPP (KKF) nearly in the same time pattern. Regardless of the discharge-depending load of suspended matter also for the nuclide load a periodicity can be verified, corresponding again especially for ^{58}Co to the inputs from KKF. At this sampling site, the ^{60}Co concentrations on average are in the same order of magnitude, although resuspension and displacement of ^{60}Co temporarily deposited on the bottom blur the correlability with the input data from KKF. The concentration changes are more distinct for ^{58}Co than for ^{60}Co; the higher ^{58}Co load is obviously attributable to the emissions of KKF, contributing in the long-term average (1978 - 1990) by approximately 77 % to the ^{58}Co contamination of the Rhine. For ^{60}Co the cumulated discharges of all Swiss installations (especially those of the Beznau NPP (KKB)) predominate against those of

KKF. For both nuclides and the same period the German shares amount to < 1 % and 4 % on average, respectively, and are therefore negligible.

To minimize the influence of resuspension and for balance computation, the corresponding annual emissions $E_{Co}(F)$ of KKF [8] as well as the cumulated emissions of Swiss ($\Sigma E_{Co}(CH)$) [6,7] and German ($\Sigma E_{Co}(D)$) installations [5] are plotted in Fig.7 for ^{58}Co and in Fig.8 for ^{60}Co and compared with the calculated annual nuclide loads L^{Kob} at Koblenz [11]. The latter were determined from the measured nuclide contents and the loads of suspended matter [12].

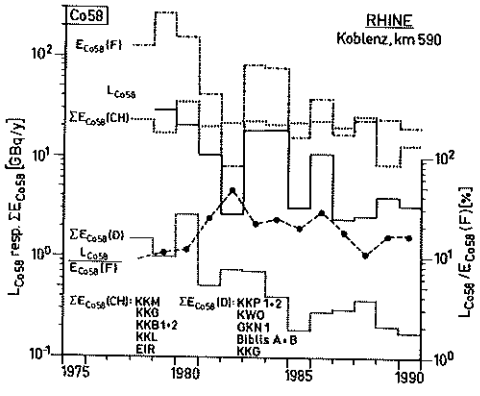

Fig.7 Annual ^{58}Co-loads L_{Co58} in suspended matter in the Rhine river at the Koblenz sampling station and emission rates for Swiss ($\Sigma E_{Co58}(CH)$) and German ($\Sigma E_{Co58}(D)$) nuclear power stations and the French NPP at Fessenheim $E_{Co58}(F)$

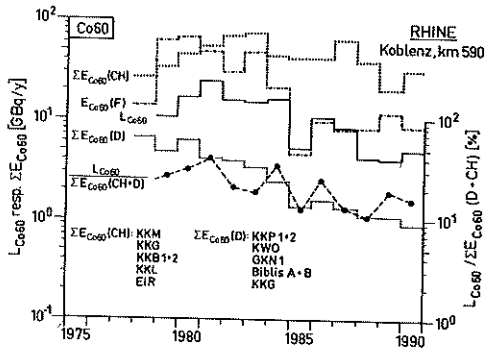

Fig.8 Annual ^{60}Co-loads L_{Co60} in suspended matter in the Rhine river at the Koblenz sampling station and emission rates for Swiss ($\Sigma E_{Co60}(CH)$) and German ($\Sigma E_{Co60}(D)$) nuclear power stations and the French NPP at Fessenheim $E_{Co60}(F)$

This comparison demonstrates that the cumulated emissions of ^{58}Co from Swiss nuclear power plants exceed those of the German installations by nearly two orders of magnitude. At the beginning of the 1980s the discharge values of KKF were higher by one order of magnitude compared with the Swiss ones but they have diminished to comparable values by 1990. The ^{58}Co loads determined at Koblenz, in turn, can be correlated quite well with the discharges from KKF, $E_{Co58}(F)$. Even the cumulated discharges of the Swiss installations, ($\Sigma E_{Co58}(CH)$), exceeding those of KKF in 1982 and 1985, were of no influence on the ^{58}Co loads determined at Koblenz. Comparative calculations show that on average (1979 - 1990), only (20 $\pm$ 10) % of the ^{58}Co released from KKF were detectable in the suspended matter at Koblenz so that the main part must have been deposited during the flow.

From Fig.8 it is evident that also for ^{60}Co the cumulated discharges from Swiss nuclear power plants, $\Sigma E_{Co60}(CH)$, always clearly exceed the nearly continuously decreasing cumulated contributions ($\Sigma E_{Co60}(D)$) originating from the German installations. The corresponding values for KKF, which were in 1980 comparable to the cumulated Swiss discharges, have remained around 30 % in general since 1985. Estimations of ^{60}Co loads at the Koblenz sampling station first showed the influence of the discharges from KKF too, but since 1987 they have also pointed at a considerable influence of discharges from Swiss installations. At Koblenz, on a long-term average (1979 - 1990), only (23 $\pm$ 10) % of the ^{60}Co activity released by all Swiss nuclear power plants and about 46 % of the amount released by KKF were detectable in suspended matter.

An assessment of the sedimentation behaviour and the elimination of the cobalt isotopes can be obtained from Figs.9 and 10. Here, the annual nuclide loads L_{Co} determined for ^{58}Co and ^{60}Co at several sampling points (Weil and Worms, Koblenz and Bockum, respectively) were related each to the corresponding cumulated Swiss discharge rates, $\Sigma E_{Co}(CH)$, at the mouth of the Aare river and

to the sums of the discharge rates from KKF including the extrapolated part f of Swiss installations, ($\Sigma E_{Co}(F) + f\,\Sigma E_{Co}(CH)$), at the Fessenheim location, respectively, and were plotted as mean values for the period 1979 - 1989 (together with the standard deviation of the single values) along the flow stretch [9]. For 1984 - 1990, at the Weil sampling site, only some 8 % of the ^{58}Co emitted from Swiss installations could be detected in suspended matter. This means that that the main part of ^{58}Co originating from Switzerland has already deposited in the Aare river and in the Upper Rhine and that the ^{58}Co found in the suspended matter at Koblenz mainly has to be traced back to effluents from the KKF. In the same period and at the same sampling point, only 14 % of the cumulated Swiss ^{60}Co effluents could be detected in suspended matter. Hence, it follows that the chief part of the Swiss effluents must also have deposited in the Aare and Upper Rhine.

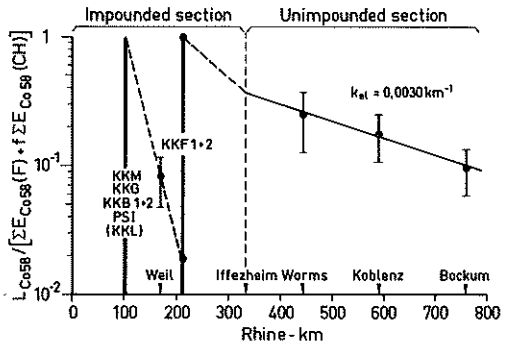

Fig.9 Plot of the average ^{58}Co-loads (interval 1979 - 1990) along the Rhine river, related to the mouth of the Aare river (sum of Swiss NPPs) and the Fessenheim NPP (KKF) location, to determine the elimination constant k_{el}

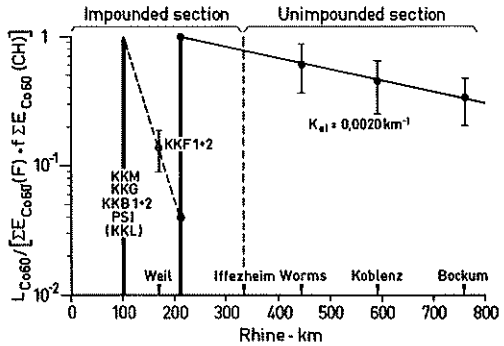

Fig.10 Plot of the average ^{60}Co-loads (interval 1979 - 1990) along the Rhine river, related to the mouth of the Aare river (sum of Swiss NPPs) and the Fessenheim NPP (KKF) location, to determine the elimination constant k_{el}

On the impounded section between the mouth of the Aare river and the Fessenheim location (10 locks), a clearly higher sedimentation rate can be expected than on the non-impounded section downstream of Iffezheim. Consequently, on the Upper Rhine stretch mean values of the elimination constant k_{el} of 0.037 km^{-1} and 0.029 km^{-1} were obtained for ^{58}Co and ^{60}Co, respectively, corresponding to a mean range of $x_R = 1/k_{el}$ = 27 km and 35 km, respectively. The sampling points downriver of Fessenheim are mainly located on the non-impounded stretch of the Rhine so that, based on the normalized loads calculated for these sites, lower elimination constants (k_{el} = 0.0030 km^{-1} for ^{58}Co and k_{el} = 0.0020 km^{-1} for ^{60}Co) and thus longer transport ranges could be determined for the Co particles accumulated on suspended matter (^{58}Co: 330 km; ^{60}Co: 500 km). On the non-impounded Rhine section as well, considerable parts of suspended matter marked with Co nuclides were deposited in sedimentation areas. Thus, at the Bockum sampling site, on average only about 10 % and 34 % of the nuclides ^{58}Co and ^{60}Co respectively, present at the Fessenheim location, could be found. By comparing the plots in Figs. 9 and 10 one finds that both nuclides differ considerably in their sedimentation behaviour: ^{60}Co shows a distinctly smaller sedimentation tendency than ^{58}Co. These differences can be explained by different states of binding of the Co nuclides in the particle matrices, resulting from different nuclear reactions during the neutron activation of different original materials [10,11].

4. Radiological assessment

Nuclear facilities release mainly ^{3}H, ^{58}Co, ^{60}Co, ^{131}I and ^{134}Cs with their wastewater, of which only ^{3}H can be measured in water with acceptable effort. The other nuclides, in principle, are only analytically detectable by applying larger technical resources (e.g. enrichment procedures).

Amongst the different exposure paths, the most sensitive one for tritium is the path "drinking-water" and for the Co isotopes the path "exposure on spoil fields", contributing to internal and external radiation exposure of man in each case. On the river Rhine the highest exposures are reached in the

Table: Hypothetical dose rates for the population in the Middle Rhine area in 1990 due to tritium-loaded tap water (consumption: 800 l/y) and exposure on contaminated sediment deposited on dumping sites (deposit thickness: 60 kg/m^2, exposure time: 1000 h/y) and utilization factors H^{eff}/H^{adm} in comparison with the admissible dose limit H^{adm} according to para. 45 of the German Radiological Protection Ordinance

Source	Nuclide	Medium	Activity concentration (Bq/kg (DM))	H^{eff} (μSv/y)	H^{eff}/H^{adm} %
Natural nuclides	^{40}K	water	0.16	0.62	0.21
	^{3}H	water	0.5	0.0064	0.002
	^{40}K	sediment	600	18	6.0
Bomb fallout	^{3}H	water	4	0.051	0.02
	^{137}Cs	sediment	8	0.95	0.32
Chernobyl NPP accident	^{134}Cs	sediment	11	3.6	1.2
	^{137}Cs	sediment	80	9.5	3.2
Nuclear facilities	^{3}H	water	5	0.064	0.02
	^{58}Co	sediment	1	0.2	0.07
	^{60}Co	sediment	2	1.0	0.33
Other tritium users	^{3}H	water	<1	<0.013	<0.004

Dose factors from [13]; DM = dry mass (for sediment)

Middle Rhine downstream of Mainz. In the table, the hypothetical radiation dose rates determined for standardized boundary conditions and the utilization factors related to the admissible dose limits for adults are assigned to several radiation sources [3,10].

From this comparison it is evident that the by far highest dose rate contribution is associated to the naturally existing ^{40}K. The dose rate contributions to the drinking-water path (^{3}H) and the spoil field path (Co isotopes), cumulated from all nuclear facilities operating in the Rhine basin are at the maximum 0.025 % and approx. 0.4 % respectively, of the dose limits associated with para. 45 of the German Radiological Protection Ordinance. Therefore, under conditions of normal operation, the input of radioactive substances from nuclear power plants located in the Rhine basin leads to negligible radiation doses on the sensitive exposure paths.

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ARTIFICIAL RADIOACTIVITY IN TEJO RIVER

MARIA CAROLINA VAZ CARREIRO
MARIA MANUELA ASSUNÇÃO SEQUEIRA
Departamento de Protecção e Segurança Radiológica
Laboratório Nacional de Engenharia e Tecnologia Industrial

Summary

The radiological survey of Tejo (Tagus) River is performed since 1975, concerning the natural radioactivity (K-40 and the natural series of the U-238 and Th-232) and the artificial radioactivity (Cs-137, Sr-90 and H-3). A radioecological study at three sampling stations has already been done (1987-1989), and the water and sediments chemical composition is still being analysed. Portugal does not have nuclear facilities, however Tejo River whatershed undergoes the discharges of radioactive liquid effluents due to the normal operation of some Spanish nuclear power plants upstream the river. Data on the artificial radioactivity in water, sediments, hydrophytes and fish, during 1990-1991, are presented in this paper. The fluctuations in the radioactivity values and the potencial radiological risks to the populations are discussed.

1. Introduction

Tejo River, originating in Spain, has been subject to radiological survey since 1975. Although there are no nuclear power plants in the Portuguese sector of the river, they do exist upstream, in Spain. Therefore the follow up of the natural and artificial radioactivity [2], [4], [7] of the river is useful to reveal any ocasional change to the normal situation, towards understanding and interpreting it. An ecological characterization of the river sampling stations [5], was accomplished between 1986-1989, and some chemical analyses are still being done, which are very helpfull in terms of radioactive data interpretation.

Three sampling stations have been retained: Vila Velha de Rodão (subsequently called Rodão), the first village in the Portuguese territory; the Reservoir of the Fratel Dam, some kilometers downstream; and Valada do Ribatejo (Valada), where there is a water captation for Lisbon supply. The studied compartments of the ecosystem are water, sediments, hydrophytes and fish.

2. Material and Methods

Radiochemical analysis of river water was carried out on 40 l, 0.45 μm filtered samples. The radiochemical analyses for Sr-90 and Cs-137 are already described [2], [4], [7]. Radioactivity measurements were performed by beta-counting in a low-background proportional gas flow counter. Tritium was measured by liquid scintillation [4]. River bank sediments after drying were subjected to quantitative gamma-spectrometry, on the whole sediment and on the fraction < 212 μm. Fish and hydrophyte species are the most common on the sampling stations, as referred in [5]. Fish muscle and hydrophytes were dry ashed and also gamma analysed. The quantitative gamma-spectrometry was performed on a Ge detector linked to a 8000 channel analyser.

The annual means presented, are geometric means and the standard deviation is the highest value out of the two standard deviation values of a geometric mean.

3. Results and Discussion

Water. Artificial radioactivity values in river water presented a certain variation along each year as shown, for exemple, in Figure 1. Most interesting is the tritium enhanced values at the end of 1991, which continued throughout the first months of 1992 (unpublished results). Similar increases of the H-3 concentrations have been noticed during 1987

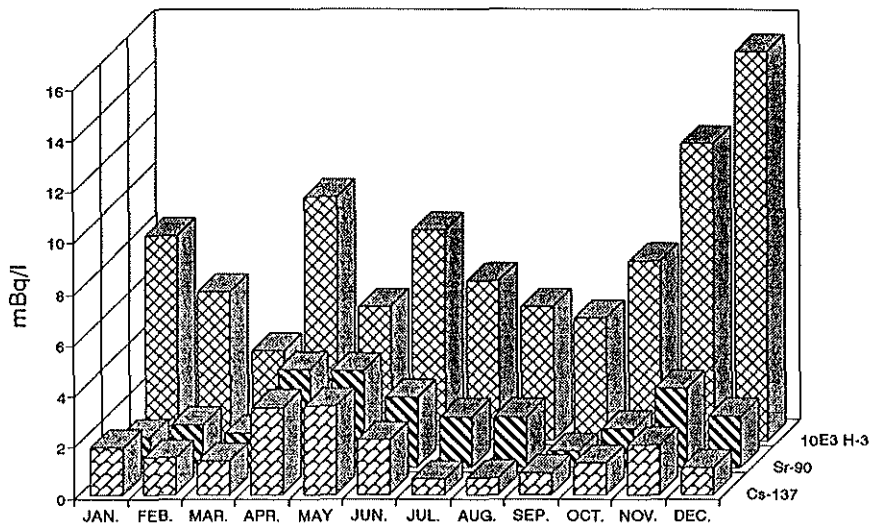

Fig.1 - Artificial Radionuclides in FRATEL Water (1991)

and 1989 [7]. At Fratel the overall 1991 mean, however, is only the double of the 1990 mean, Table I, and it is merely slightly higher than the values recorded between 1977 and 1986, which varied from 1.5 to 3.0 Bq l^{-1} [7].
It is also interesting to compare H-3 concentrations in river and rain water at Fratel, where it only may be imputed to worldwide fallout. The annual means in 1990 and 1991 are respectively 0.64 ± 0.18 and 1.1 ± 0.2 Bq l^{-1}. H-3 concentrations in rain water are referred in the literature such as: 1.4 to 4.6 Bq l^{-1} during 1988-89 in Danemark [1] and 0.4 to 3.7 Bq l^{-1}, during Oct.-Dec. 1990 in USA [6]. This last report presents for the same period in surface waters, values from 0.4 to 9.6 Bq l^{-1}. In Rôhne River [3] upstream Marcoule (reprocessing plant), but already downstream some nuclear power plants, H-3 concentrations vary between 2 and 18 Bq l^{-1}.

Table I - Artificial radionuclide concentrations in Tejo River water (annual means)

Year	Sampling Station	Dissolved matter (0.45μm) Cs-137 mBq l^{-1}	Sr-90 mBq l^{-1}	H-3 Bq l^{-1}
1990	Rodão	1.5±0.4	1.9±0.4	3.9±1.5
	Fratel	2.0±0.4	1.9±0.3	3.3±1.2
	Valada	1.2±0.7	1.0±0.5	3.0±1.3
1991	Rodão	1.2±0.5	3.0±0.5	6.5±0.9
	Fratel	1.4±0.3	1.9±0.3	6.9±0.9
	Valada	1.1±0.5	1.9±0.1	4.6±0.9

Concerning Cs-137 and Sr-90, Table I, the annual means are very similar to the preceeding years, excepting 1987 where enhanced concentrations were measured, mainly for Cs-137, with annual means of 9.0 ± 4.9 mBq l^{-1} and 6.3 ± 2.1 mBq l^{-1} respectively at Rodão and Fratel [7]. Cs-137 and Sr-90 concentrations in river water are comparable with rain water, where the annual means, in 1990 and 1991, are respectively: 0.76 ± 0.19 and 1.3 ± 0.3 mBq l^{-1} for Cs-137; 0.87 ± 0.17 and 0.91 ± 0.19 mBq l^{-1} for Sr-90.

Sediments. We may see in Table II that the fraction < 212 μm has higher values than the whole sediment, being more representative, specially when normalized to the aluminium content. Actually, aluminium, the main component of clay minerals, which concentrate radionuclides most extensively, must be in the finer fraction. This fraction represents, in mean, 34%, 5.1% and 6.7% of the whole sediment, respectively at Rodão, Fratel and Valada.

Table II - Cs-137 concentration in Tejo River sediments (annual means)

Year	Sampling Station	Cs-137 Bq kg^{-1} (dry) Sediment total	Fraction <212μm	Bq Cs-137/g Al Sediment total	Fraction <212μm
1990	Rodão	4.3±1.1	6.2±1.5	0.42±0.11	0.61±0.15
	Fratel	2.7±1.5*	5.2±2.8*	0.29±0.16*	0.57±0.30*
	Valada	3.4±2.4*	3.7±0.9	0.41±0.29*	0.45±0.11
1991	Rodão	5.2±1.6	6.8±2.0	0.51±0.16	0.66±0.20
	Fratel	1.3±0.3	3.6±0.2	0.14±0.03	0.39±0.02
	Valada	1.4±0.4	3.0±0.7	0.17±0.05	0.36±0.08

* the upper value of the standard deviation is above 50% of the GM

Hydrophytes and Fish. Cs-137 concentrations in Hydrophytes and Fish muscle are low, Table III, but in agreement with the values generally observed in Tejo River [7]. As it is also usual, Be-7 (cosmic radionuclide) is almost always measured in hydrophytes, showing however a wide range of values.

Table III - Cs-137 concentration in Fish and Hydrophytes from Tejo River (annual means)

Year	Sampling Station	Fish muscle Bq kg^{-1} (wet) Cs-137	Hydrophytes Bq kg^{-1} (dry) Cs-137	Be-7
1990	Rodão	—	3.4±1.0	16±5
	Fratel	0.13±0.08*	≤1	7±3
	Valada	0.21±0.17*	≤1	15±7**
1991	Rodão	0.24±0.13	1.7±0.8	8±4**
	Fratel	0.09±0.03	3.8±1.9	29±12
	Valada	—	—	—

* the upper value of the standard deviation is above 50% of the GM ** only one value

Thereafter, an assessment of the radioactivity annual intake and of the annual dose for an eventual critical group, local fisherman, could be done. Taking into account the main pathways, water for H-3 and Sr-90, water and fish muscle for Cs-137, a conservative hypothesis was considered: direct intake of river water without treatment processes, 2 l/day, freshwater fish consumption of 300 g/day and the highest values of the referred radionuclides measured in the period 1990-1991 at Fratel (Fig.1). Therefore, the evaluated annual ingestion is maximized: 1.1 X 10^4 Bq of H-3, 61 Bq of Cs-137 and 3 Bq of Sr-90. However the estimated values are low compared with the ALIs for the referred radionuclides.

The annual effective doses estimated, using dose factors [8], are also low: 0.19 μSv for H-3, 0.73 μSv for Cs-137 and 0.11 μSv for Sr-90.

4. Conclusions

Regarding data obtained it is possible to infer that the tritium changes in river water might be imputed to the normal operation of the spanish nuclear power plants.

Sediments show to be the ecosystem compartement, in Tejo River, that preferentially accumulates artificial radioactivity.

The estimated annual intakes are very low compared with ALI, and the doses to the public arising from the assumed ingestion from the radioecological point of view are not significative.

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ENVIRONMENTAL MONITORING WITH A STEPWISE ROTATED AEROSOL FILTER SYSTEM

K. Heinemann, K.H. Schneider, R. Hille
Abteilung Sicherheit und Strahlenschutz
Forschungszentrum Jülich GmbH, Postfach 1913, D-5170 Jülich

Summary

A newly developed and tested stepwise rotated aerosol filter system is described. Three end-window detectors measure the cross beta activity : one during sampling time, one immediately after sampling, and one after the decay of the short-lived natural activity. The measuring range reaches from 3 10^{-4} to more than 10^{6} Bq/m³. Therefore this device is suitable for environmental aerosol monitoring of nuclear plants, both during normal operation and in case of an accident. Some special software tools indicate immediately a deviation from normal operation.

1. *Indroduction*

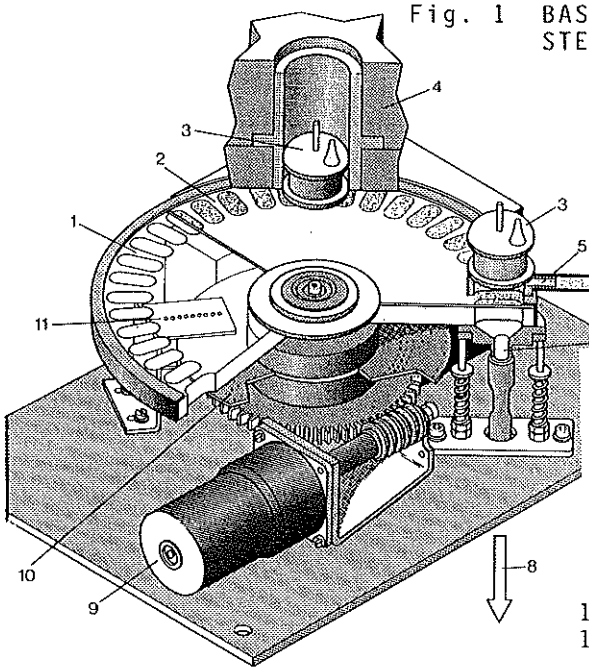
At the Research Center Jülich a variety of different sources (research reactor, hot cells, accellerators, medical installations, and waste utilies) release radioactive materials into the environment, into the atmosphere and into the surface water. Therefore the environmental surveillance has to be comprehensiv and unnormal radiological situations must be indicated immediately.

At the Research Center Jülich environmental monitoring includes on the one hand a discontinous control system consisting of the analysis of many different samples in the laboratories and on the other hand a continuous surveillance which is performed by ODL (local dose rate) measurements and by cross beta aerosol activity determination in the so-called "inner surveillance circle". This circle consists of 8 monitoring stations near the border of the center site. After more than 30 years successful operation these stations must be modernized. Therefore the measuring devices were reconstructed according to state of the art. In this frame the stepwise moved filter tape which was in use for more than 20 years for the surveillance of radioactive aerosols will be replaced by a well tested /1/, /2/, stepwise rotated aerosol filter system with digital evaluation of all measurements.

2. *Basic construction and specifications*

Fig. 1 shows the basic construction of the filter system. The air, coming from the right side, passes with 3 m³/h through a glass-fiber filter (Schleicher und Schüll Nr. 10) which is supported by a perforated metalic disc of 380 mm diameter. The perforation consists of 43 spots with the dimensions of 6 x 20 mm², 41 of these spots are to measure the air contamination, one spot is to calibrate the detectors (source: 10 Bq Sr 90/Y 90), and one to determine the background rate. Three end-window detectors (LND-7242) measure the contaminated filter spots: detector D1 during sampling time, detector D2 immediately after sampling and detector D3 after the decay of the short-lived natural activity five days after sampling. Each detector has the same integral counting time of 24 hours. To reduce the background

Fig. 1 BASIC CONSTRUCTION OF THE STEPWISE ROTATED FILTER SYSTEM



1 rotated disc with holes
2 filter
3 end-window detector
4 shielding
5 contamination chamber
6 air input
7 suck-in device
8 to the pump
9 motor to rotate the disc
10 coded position disc
11 position control

radiation each detector is shielded by 50 mm lead. Because of the diameter of the detectors and their shieldings the filter has to be rotated in each step more than the angle difference between two neighbouring spots. Therefore each filter is rotated more than 360^0 to use every possible filter position. The positioning is exactly controlled by an optoelectronical gray code decoder (MEGATRON 390G/4096/5/CDIR). Under normal conditions the aerosol concentration can be monitored for longer than one month. To guarantee a surveillance without interruption two such stepwise rotated filter divices form one unit. While the last spots of the first device are still being measured by the detectors D2 and D3, on the second device air sampling already starts.

The detection limit of better than $3\ 10^{-4}$ Bq/m³ for the long-lived activity measured by detector D3 is calculated /3/ from the background pulse rate (0,1 pulses per second), the efficiency of the detector (0,15) and the normal sampling and counting time of 24 hours.

In case of an accident the sampling and counting time will be automatically reduced according to the counting rate and the aerosol activity will be measured only by detector D1, because in this case the concentration of the natural radioactivity can be neglected. The upper detectible aerosol concentration in air is determined from the sampling time of 1 minute, the efficiency of 0,15 and the maximal counting rate of 10 000 pulses per second. It is less than $2\ 10^6$ Bq/m³. If the tube is also operated in the non-linear part of its characteristic the maximum measurable concentration increases by a factor of about 6. Thus this device is particular suitable for aerosol monitoring of a nuclear plant, both during normal operation and in case of an accident.

The pulse rate of each detector is counted in time steps of 10 minutes. To get an early information of a deviation from normal operation the time depending curve measured by detector D1 is continuously compared with the upper uncertainty range of curves derived from other measurements of detector D1.

The detector D2 measures the decay pattern from the radioactivity of the contaminated filter spot. By fitting this measured curve with the naturally occuring radioactivity of the nuclides Pb 212 and Pb 214 it is possible to detect even a low amount of unknown artificial radioactivity within about 20 hours after the end of samplimg time. This is much shorter than usual: The normal operation procedure where a filter is contaminated for about two weeks and is measured in the laboratory after the decay of the short-lived natural activity needs much more time. Under the most unfavourable conditions the first information of an unnormal situation in the environment is obtained about 20 days after the event without any indication of the exact day of activity release.

The fitting of the measured curve is done according the following equation:

$$F(t) = a_1 \exp(-\lambda_1 t) + a_2 \exp(-\lambda_2 t) + a_3 \exp(-\lambda_3 t) + b$$

λ_1 and λ_2 are the known decay probabilities of the nuclides Pb 212 and Pb 214. All other parameters have to be fitted. The parameter b refers to a long-lived artificial radioactivity or a change in the background rate. a_3 gives the amount of a short-lived unkown nuclide at the beginning of the computer analysis and λ_3 is its decay probability.

The indication of an artificial radioactivity by the computer analysis has to be verified by detector D3 later on.

In case a certain amount of artificial radioactivity has been detected, the filter spot will be analysed again in the laboratory e.g. by gammaspectrometry.

Typical curves of the detectors D1 and D2 and their computer analysises are shown in Fig. 2 and Fig. 3. Fig. 2 presents the time dependent measurements of detector D1 (full line) and the upper range of uncertainty (dotted line). This is the maximum value which should not be exceeded by the measured curve at a certain time under normal conditions. In Fig. 3 the time dependent measurements of detector D2 are given (unregular full line). The other soft full curve is the computer analysis of the decay pattern. The contributions of the nuclides Pb 212 and Pb 214 are presented. In the example a long-lived artificial contribution is shown which has been simulated by a weak Cs 137 source. The detector D3, as a rule, measures constant values. Therefore no example for such a curve is given here.

3. *The central control unit*

In the modernized monitoring stations of the inner surveillance circle in the Research Center Jülich a new ODL system is also applied. It is the "Janus-head" system which has a measuring range from background up to 30 Sv/h. Its response does not depend on the direction of the radiation field. It has been described in more detail in /4/.

The stepwise rotated aerosol filter unit and one ODL device are placed in a container and form the minimal equipment of the stations of the inner surveillance circle. In some stations sampling devices for aerosols and iodine are additionally

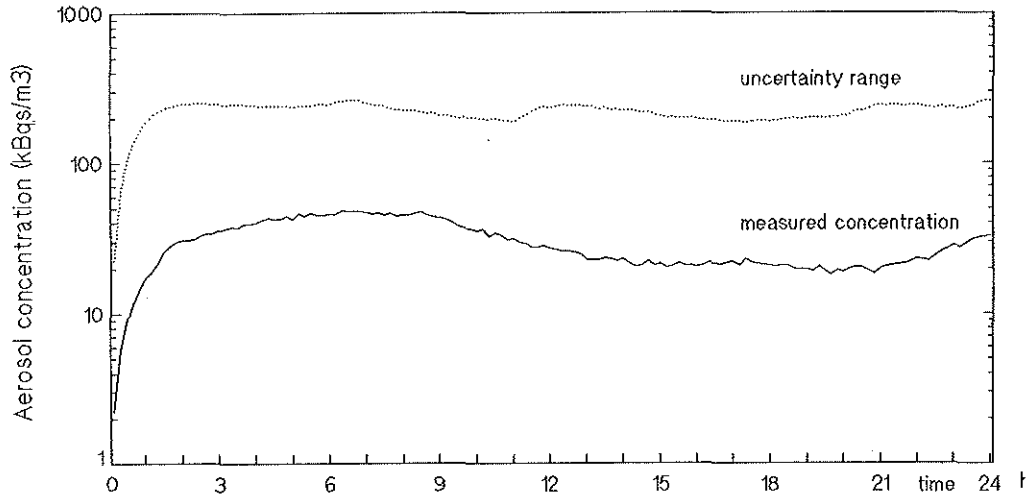


Fig.2 Radioactivity concentration measured by detector D1

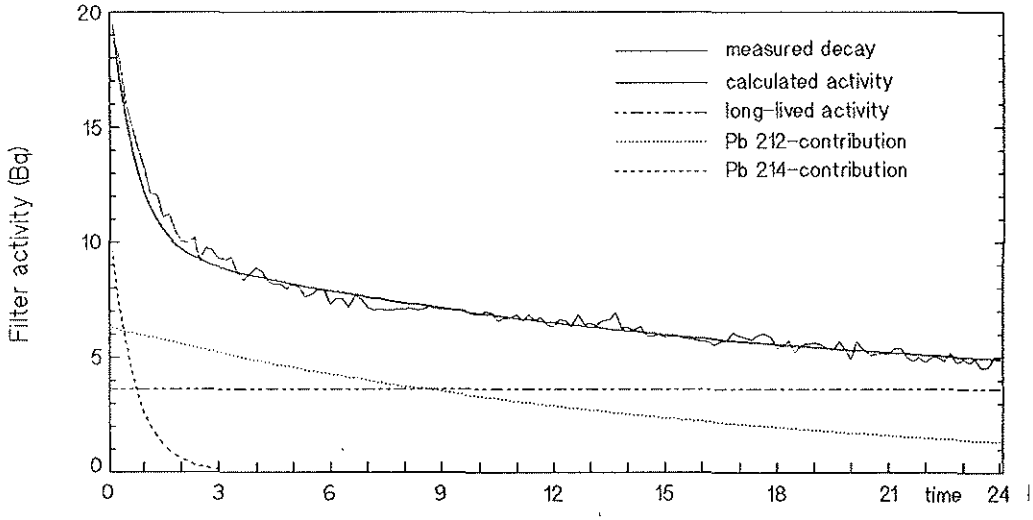


Fig.3 Decay analysis detector D2

installed. This year a new station (nr. 8) had to be equipped only with a gamma and a neutron ODL (BIOREM FAHT 750) system for monitoring the radiation outside a storage for spent high temperature fuel elements.

In case of an interruption of the power supply or a break down of the data lines to the central computer unit all data are saved in the stations for about two days.

All original data of a station are firstly handled by a front end processor in the station, which works as a stand alone computer. From the front end processor all data of the eight stations of the inner surveillance circle are transferred to a central computer unit. The main tasks of this central computer unit are:

- to control the operations of all connected equipments
- to store and analyse the data
- to present the time dependence of selected measurements on a monitor and
- to alarm in the case of systems' failure or if the activity limits are exceeded.

All actual meteorological parameters important for the interpretation of the activity data are also available in the central computer unit. In the case of an increased radioactivity at a certain station the number of possible radioactivity emitters may be reduced by knowledge of the wind direction and viceversa.

Each change in the adjustance of parameters is printed by the central computer unit. About each month the data stored are backed up and transferred to a floppy disc.

The reconstruction of the inner surveillance circle will need several years and will end with the introduction of a second computer unit for redundance reasons.

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USE OF AL_2O_3:C FOR ENVIRONMENTAL MONITORING

K. Gmür and C.Wernli
Paul Scherrer Institut
CH-5232 Villigen PSI, Switzerland

Summary

Al_2O_3:C thermoluminescence dosemeters have been tested for one and a half year. Sensitivity, reproducibility of measurements, and fading characteristics have been compared to CaF_2:Dy which is routinely employed for environmental monitoring, and the photon energy response using a compensating filter was measured. Special emphasis was given to the determination of parameters which enable automatic read-out and annealing of the TLDs in a hot-gas reader normally employed for personal dosimetry. The results are encouraging enough to start the replacement of CaF_2:Dy by the new material.

1. Introduction

Several efforts were made to improve the existing environmental monitoring system based on CaF_2:Dy and spherical energy compensating filters. The method involves time-consuming manual labour and is susceptible to errors. CaF_2:Dy is not re-usable after read-out, but has to undergo an annealing process at 400°C. After having tested various new TL materials for their suitability in environmental monitoring we conclude that anion defective Al_2O_3:C [1,2], with its photon effective atomic number of 10.2 being closer to that of tissue than CaF_2, meets the requirements.

2. Materials and Methods

We are using Al_2O_3:C pellets having a diameter of 5 mm and a thickness of 1 mm, obtained in two batches from Victoreen (Model 2600-80). The TLDs were given a thermal treatement at a temperature of 400°C for 1 hour and a cooling time of 24 hours. Then the pellets were filled into slides of the Dosacus reader from Alnor, normally holding 4.5-mm-diameter LiF pellets. Thus, the slides had to be modified slightly. The cartridges were placed into the dosemeter badge normally used for personal dosimetry.

The sensitivity factors of the TLDs were individually determined and stored. A difference of 20% was found between TLDs from batch one and batch two. The Dosacus hot-gas reader operating at 280°C and with a heating time of 12 seconds serves to measure the TL signal. The same heating cycle is applied two or three consecutive times to regenerate the dosemeters immediately before exposure.

3. Results

Although the above mentioned reading cycle with its high heating rate of some 25°C/s is not optimal for best performance, the residual background dose is 2 ± 0.5 μSv, including the reader's inherent signal of ca. 0.5 μSv. The uncertainty is about half the value of a CaF_2(Dy) chip.

Reproducibility was tested by irradiating the TLDs with 1 mSv and correcting the responses with the individual sensitivity factors. The standard deviation of the results was ±2% per pellet.

The response of the dosemeter after irradiating with 2 mSv ambient dose equivalent $H^*(10)$ at various energies is shown in Figure 1. The energy compensating filter is the same as used for personal dosimetry with LiF, namely metallic aluminium of 2 mm thickness incorporated in the badge. Experimental points are normalised at 662 keV photon energy. A small over-sensitivity at 50 keV photon energy is counterbalanced by the undershoot around 150 keV. The energy-dependent response meets the criteria given in ANSI N545-1975.

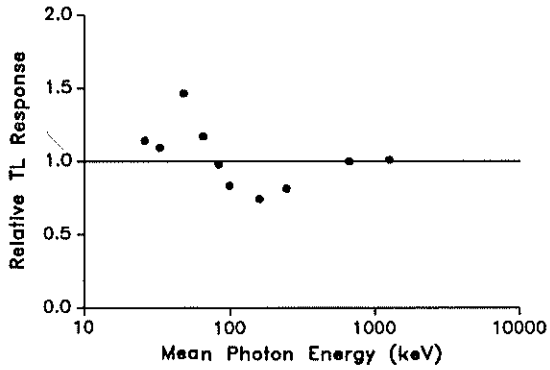

Figure 1: Photon energy response of filtered TLDs having received 2 mSv ambient equivalent dose $H^*(10)$.

Figure 2 displays the result of an 8-month measurement of the fading. The experimental points are mean results of 4 TLDs for storage times up to 4 months, and 8 TLDs for longer storage. They are corrected for natural background contribution and normalised at unity. The uncertainties are standard deviations of the mean. All detectors were set at their background level on December 13, 1991. The subsequent radiation doses were 1 mSv from ^{137}Cs, the first one applied on December 16, 1991, and the last one a day before read-out on August 21, 1992. The dosemeters were stored in darkness at room temperature in an office. The natural background dose was 0.366 mSv. With these laboratory conditions the fading is less than 2% per 8 months. Additional field fading tests were conducted, and the results show fading rates of less than 2% per 3 months. These low fade rates are in contradiction to earlier measurements, when a signal loss of 15% within the first 10 days of storage had been observed. It was not possible to reproduce the result. Other investigators also report on conflicting fading measurements [3].

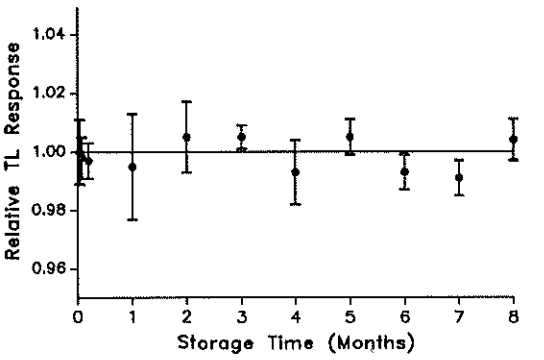

Figure 2: Fading of Al_2O_3:C at ambient room temperature.

A separate set of TLDs was regenerated on August 14, 1992. They received radiation doses on August 19 and 20 and were evaluated on August 21, together with the dosemeters of the long-term study. The responses, corrected for the appropriate background, did not differ from those of the dosemeters regenerated in December 1991.

Light-induced fading is not negligible. Bright day light or intense light from a fluorescent lamp produce substantial loss of stored signal even after minutes, if the TLDs are not covered [4]. Optical fading was also found with TLDs which were encased in the dosemeter badge and exposed to day light for several days. As a counter-measure the badges for environmental monitoring were equipped with black paper light shieldings.

A direct comparison with CaF_2:Dy dosemeters placed at various locations near nuclear power plants was carried out. The measured doses ($H^*(10)$ for Al_2O_3:C, H_x for CaF_2:Dy) range from 0.13 mSv to 1.3 mSv per time period of 3 months. In general, the Al_2O_3:C TLDs yield figures which are 5% lower than those from CaF_2:Dy. It is supposed that the different energy filtering or the non-isotropic sensitivity of the dosemeter badge contribute to the difference.

4. Conclusion

The usage of Al_2O_3:C thermoluminescent material for environmental monitoring is advantageous, both because of its excellent radiological properties and the possibility of automated operation. The closeness of the effective atomic number and of the diameter of the chips to those of LiF renders the possibility to work with dosemeter badges and reader used for personal dosimetry, saving a considerable amount of man-power without loss of quality. No essential thermal fading was observed. However, the TLDs are light sensitive and have to be carefully protected from incandescent light. The gas temperature of 280°C is nominal only. Some adjustement may be needed, depending on heating time and gas flow. If the temperature is too high, it is not possible to reset the TLDs at their background level of 2 μSv. Minor mechanical modifications have possibly to be made on the reader, because of the larger diameter of the pellets.

Acknowledgements

We wish to express our thanks to Martina Merk and Arnold Finsterwald for carefully carrying out the measurements.

References

[1] M. S. Akselrod, V. S. Kortov, D. J. Kravetsky and V. I. Gotlib: "Highly Sensitive Thermoluminescent Anion-Defective α-Al_2O_3:C Single Crystal Detectors"
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[2] M. S. Akselrod, V. S. Kortov and E. A. Gorelova: "Preparation and Properties of α-Al_2O_3:C"
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[3] R. E. Zelac and A. Jones: "Suitability of Al_2O_3:C for Environmental and Other Low-Level Radiation Monitoring Based on Temperature Dependent Fade Rate Characteristics"
Private communication

[4] D. A. Levy and R. E. Zelac: "Suitability of Al_2O_3:C for Environmental and Other Low-Level Radiation Monitoring Based on Results of Several Laboratory Tests"
Private communication

AREA RADIATION MONITORING SYSTEM: THE AAM-90 MONITORING SYSTEM

B. Reichert, M. Schütz
Herfurth GmbH, Hamburg

The Modern Doserate Monitoring

The Alnor AAM-90 is an environment monitoring system with substantial application possibilities for measuring gamma and X-ray radiation.

It is mainly used to monitor the environment in and around nuclear technical plants, hospitals or wherever it is necessary to monitor x-ray and/or gamma radiation (e.g. as level sensor). The scale ranges from one probe system to a multiple probe system, from the connection of a portable doserate measuring instrument up to PC controlled data processing.

Basic System Component:

The core of the system is the intelligent Probe RD-02. Based on two GM tubes (low and high dose rate) it comprises a large doserate measuring range, beginning at natural background radiation up to catastrophically high levels. The operating temperature range lies between -40 °C and +70 °C. Enclosure classification is IP67, which means the probe is water resistant and can be used at almost all conditions, e. g. for use outdoors. Power supply is made through probe cable. An integrated backup-accumulator allows for longterm continuous operation. Up to 864 measuring results can be registered and stored, and, if necessary be called up through an RS-232 serial interface. The intelligence of the probe allows for constant self-monitoring. This means, fault functions are also transmitted to the connected PC.

Economic Efficiency and Flexibility:

To ensure the economic efficiency of the system, it was considered to make use of already existing cables and telephone lines. Furthermore, it is possible to connect the probe to the central monitoring computer (PC) via radio (modems). As already mentioned, the probe can be directly connected to the doserate measuring unit RDS 120. This allows for a local display unit, also for checking the probe itself for proper functioning.

System Software:

A special system software AAM-90/95 allows for the setup and monitoring of as many measuring stations as the computer capacity can take. As shown on the video monitor following details can be displayed:

* doserate progress (graphic) with individual time scheme (minutes, hours, days, weeks, months and year)
* Display of as many as nine doserate alarm setpoints in 0,01 µSv/h steps
* Alarm exceedings with date, time and doserate
* Function message of probe (active/passive)

* Average doserate, averaged over 24 hours and the last minute
* General overview of query and transmittal times

Further display possibilities are only listed schematically:

* Probe parameters
* Computer-specific statements
* Fault reports
* Further: map of the stations, graphical displays of the regions to be monitored, etc.

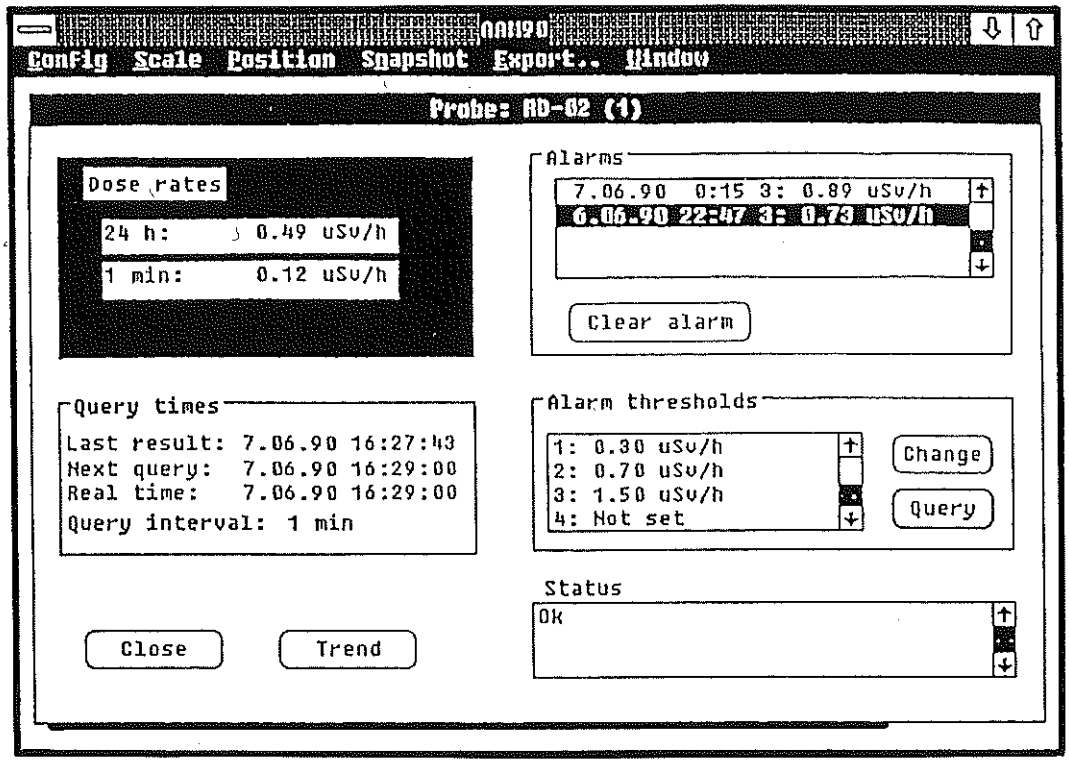
Conclusion:

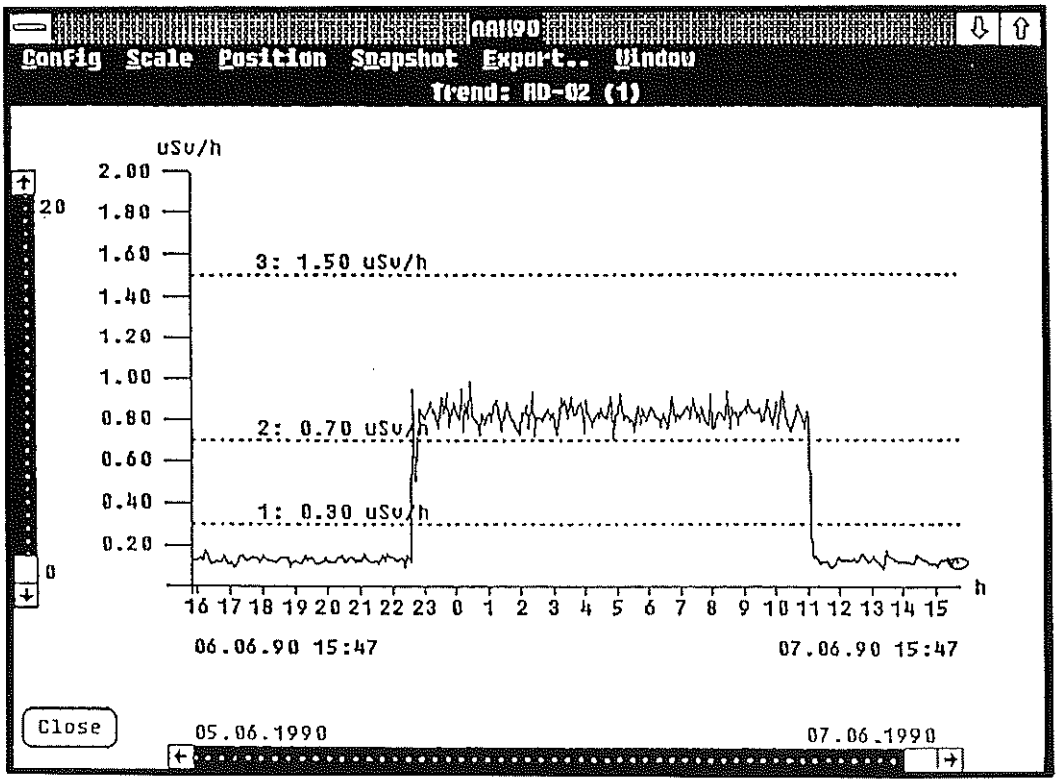
The present AAM-90 environmental radiation monitoring system is only the youngest product of a long chain of developments, beginning at the early sixties. In the meantime, the Finnish measuring network alone consists of several hundred stations. It has been applied for type approval tests for official calibration according to the test regulations for radiation protection.

Literature:

[1] Alnor, Rados Technology: Short description of the AAM-90 Area Monitoring System

[2] Alnor, Rados Technology: Leaflet "AAM-90 Area Radiation Monitoring System"


Config Scale Position Snapshot Export.. Window
Probe: RD-02 (1)
Dose rates
24 h: 0.49 uSv/h
1 min: 0.12 uSv/h
Alarms
7.06.90 0:15 3: 0.89 uSv/h
6.06.90 22:47 3: 0.73 uSv/h
Clear alarm
Query times
Last result: 7.06.90 16:27:43
Next query: 7.06.90 16:29:00
Real time: 7.06.90 16:29:00
Query interval: 1 min
Alarm thresholds
1: 0.30 uSv/h
2: 0.70 uSv/h
3: 1.50 uSv/h
4: Not set
Change
Query
Status
Ok
Close
Trend


Config Scale Position Snapshot Export.. Window
Trend: RD-02 (1)
uSv/h
2.00
1.80
1.60
1.40
1.20
1.00
0.80
0.60
0.40
0.20
3: 1.50 uSv/h
2: 0.70 uSv/h
1: 0.30 uSv/h
16 17 18 19 20 21 22 23 0 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15
h
06.06.90 15:47
07.06.90 15:47
Close
05.06.1990
07.06.1990

AUTONOMOUS DATA LOGGING (ADL): A NEW CONCEPT FOR HIGH PRECISION MONITORING OF ENVIRONMENTAL GAMMA DOSE RATE

Volker Genrich
Genitron Instruments GmbH, Heerstr. 149, D-6000 Frankfurt/Main 90

Summary

The author is going to present his findings on continuous data logging by means of weathertight, small and inexpensive devices, called ADL-monitors (ADL = Autonomous Data Logging). This new kind of equipment has been specially designed for the operation under severe conditions and can be placed anywhere in the open air. By using latest semiconductor technologies, ADL-probes operate continuously for more than 50 000 hours (5 years) without changing battery. Data can be read from the ADL-devices at any time (over a considerable distance) by activating optical (infrared) data transmission. Thus, all information is transferred into a data base on a PC-laptop computer. Profiles of dose rate over time can be examined in detail with special software tools.

1. The basic idea of the ADL-concept

For continuous registration of the environmental gamma dose rate, there are sophisticated on-line measuring networks available in many places. Additionally - in much greater numbers - passive-integrating dosimeters (e.g. TLD type) are being used, mainly for complementing active networks and for emergency dosimetry.

However, there are still a lot of tasks, where the "simple" dose-integral offered by passive detectors does not reveal enough information, while on-line measurement and data-transmission in real-time is prohibited by cost or by problematic infrastructure of the region.

The new technology as described here, is a flexible and cost-effective tool: A network equipped with ADL-probes can provide the profile of dose rate over time with a high degree of precision. Therefore, the ADL-technology may be used as an alternative or as a supplement to the well established technologies (see table 1).

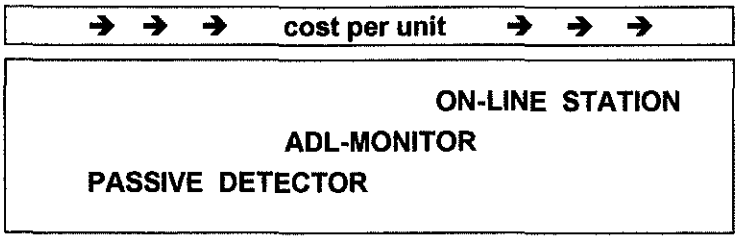


Table 1: *Complementary technologies for monitoring gamma dose rate*

A chain or circle of well-positioned ADL-probes in the environment will deliver the exact pattern of gamma dose rate on an hour-by-hour basis. Located close to a plant, a circle of ADL-probes can be used for tracing emissions. Thus volume, time history and direction can be assigned to every significant release of gamma emitters.

Furthermore, a number of ADL-probes equally distributed over a certain region, will provide a realistic data basis for gamma submersion - under natural as well as under emergency conditions. The main fields of application are summarized below in table 2 :

- ❑ investigation on radiation background **before** putting a plant into **operation**
- ❑ accurate verification of **annual dose limits** in the vicinity of a plant
- ❑ intensified plant monitoring program under **revision or fuel exchange**
- ❑ consolidating existing networks with regard to **accidental** releases
- ❑ ADL-monitors kept in reserve **to be placed by helicopter** in emergencies
- ❑ third-party **safeguard** programmes (local or federal institutions, **IAEA**)
- ❑ environmental monitoring in regions with **problematic infrastructure** (CIS)

Table 2 : *General fields of application for ADL-monitoring*

2. Scope of research and development

The author has been involved in the development of a new type of dosimeter using latest semiconductor technologies (ASIC-chip design). The first instrument was a hand-held dose rate meter called "MIRA-661", which has been designed according to a list of specifications by the Commission of the European Communities (CEC). A large number of this type has been submitted as a CEC-donation to the regions affected by the Chernobyl accident (Ukraine, Belorussia and Russia).

The hardware for ADL-monitor is based on the experience of the MIRA-project. The following objectives (table 3) have been set up in the beginning of 1992 for an ADL-project called *"GammaTRACER"*:

- ❑ rugged construction with extremely high **reliability** (using redundancy etc.)
- ❑ optimised power management for **several years of continuous operation**
- ❑ data storage in memory every hour, **history available for 6 months**
- ❑ **optical data link** to PC-laptop computer over a distance of 5 meters
- ❑ file management on HD-drive, **data bank** for up to 256 ADL-monitors
- ❑ evaluation of dose rate profiles with **software tools** under *"MS-WINDOWS"*

Table 3 : *Objectives for the GammaTRACER project (Autonomous Data Logger)*

3. Field-tests with a prototype of the ADL-monitor "GammaTRACER"

The following figures and diagrams represent some typical results from ADL-monitoring as so far obtained.

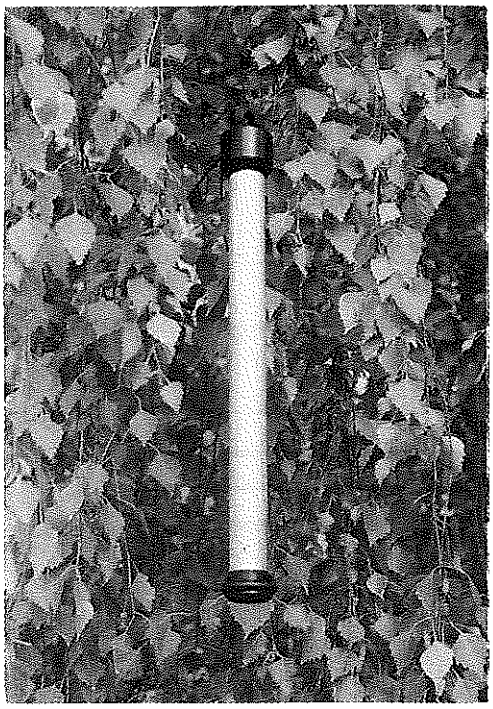


Figure 1 :
Fixed at the top of a tree - a typical site for an ADL-monitor in the open air

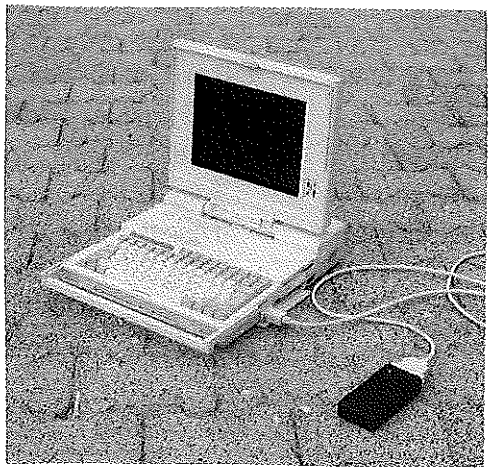


Figure 2 :
Mobile equipment for reading ADL-data : PC with IR-optical transceiver (right)

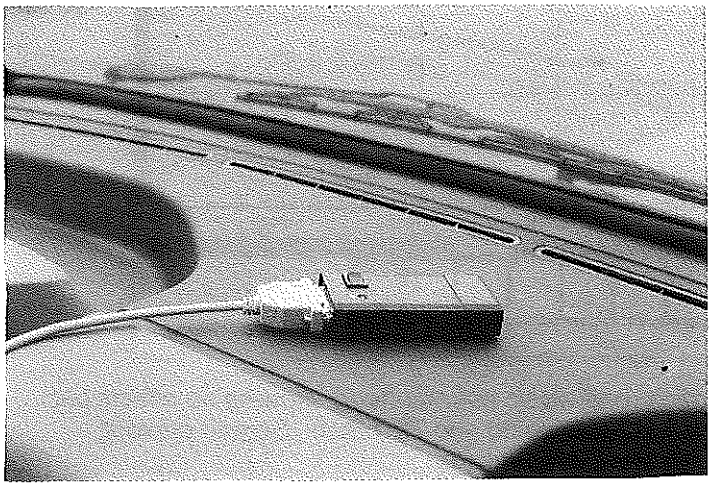


Figure 3 :
To minimise personal radiation exposure in a contaminated area, data transmission can be accomplished from inside a car within 30 seconds (transceiver operating through the car's windshield)

Detector:	2 separate Geiger-Müller-tubes (redundancy)
Energy range:	42 - 1300 keV +/- 30 %
Sensitivity:	60 cpm at 100 nGy/h
Range:	A) option "S" 10 nGy/h - 30 mGy/h
	B) option "H" 10 nGy/h - 10 Gy/h
Cycle time:	10 min or 1 hour
Case:	metal pipe, sealed hermetically (submersible to 10 m depth)
Dimension:	diameter 40 mm, total length 550 mm
Weight:	1.2 kg
Battery life:	50 000 h (5 years)

Table 4 : *Technical data of the autonomous data logger "GammaTRACER"*

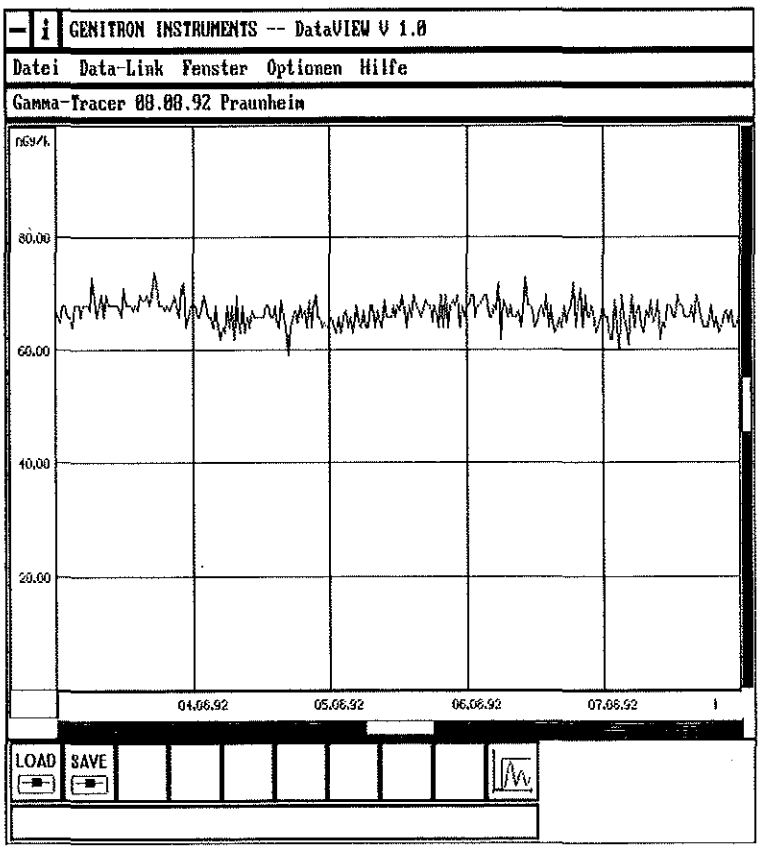


Figure 4 : *Dose rate recording (one week, hour-by-hour) showing no effects but stochastic fluctuation between 60 and 72 nSv/h. Screen printout, performed with "DataVIEW" software.*

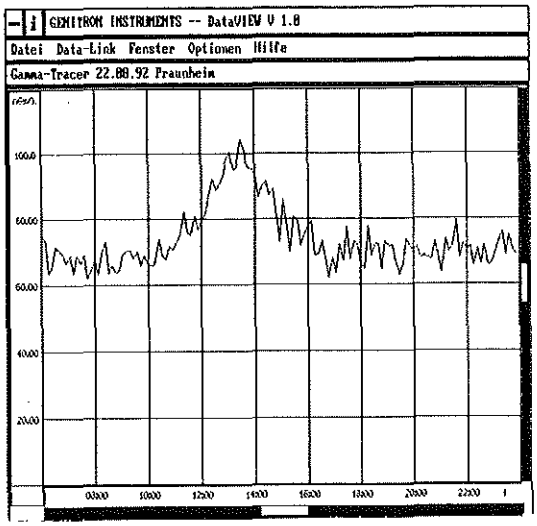


Figure 5 :
Dose rate recording (10 min intervals for 12 hours) showing a definite elevation from 65 to 100 nSv/h, caused by natural wash-out phenomena (radon pro-geny). Screen printout, performed with "DataVIEW" software.

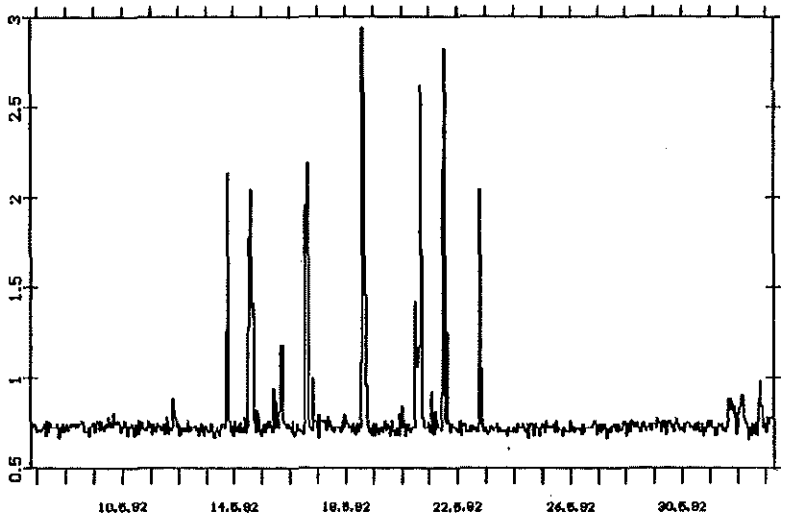


Figure 6 : *Dose rate recording from May 07 until June 03, showing different spikes due to the operation of an accelerator facility (GSI, Darmstadt)*

4. Discussion and further prospects

Results from ADL-monitoring demonstrate, that these devices are well-suited for tracing minor fluctuations in the environmental gamma dose rate. They are quite easy to install, as they can be put into operation at any place in the open air within a a minimum of time.

The author would like to invite all radiation protection specialists to look for an application of the ADL-concept and to take part in a field test - free of charge. Please don't hesitate to contact the author.

INTELLIGENT DOSE RATE DETECTORS FOR RADIATION MONITORING IN AUTOMATIC MEASURING NETWORKS

W.Rieck, H.Faatz, J.Frowein, M.Iwatschenko-Borho, H.Schleicher
FAG Kugelfischer Georg Schäfer KGaA
Radiation Measuring Technology Devision
POB 1660, W-8520 Erlangen, Germany

Summary

Several doserate detectors for gamma and neutron radiation have been specially designed for the application in networks: All the electronic components which are necessary for an automatic measurement mode are integrated inside the detector. This also includes a microprocessor and a serial interface. On request the probe transmits the complete measurement information. Up to 32 probes can be operated in one local network.
A newly developed PC program (MRP = Monitoring Radiation Program) allows to handle, display, and store the data of up to 32 intelligent probes additionally to the data of other radiation monitors.

Introduction

Automatic ambient radiation measurements require the periodic acquisition and processing of measured data. For this purpose in conventional systems the detector signals have to be amplified, computed and translated into a computer readable format. The new FHZ 601 A intelligent dose rate detectors perform all these steps of signal processing itself by its integrated microprocessor and analog devices. The detector provides complete data sets, calibrated and properly checked for any errors. They are available via a serial interface.

1. Detector

The FHZ 601 A intelligent probe combines the adventages of modern microprocessor electronics with the excellent characteristics of the FHZ 600 A proportional detector /1/ which is successfully in use in several measurement networks and monitoring stations. The high sensitivity and the wide measuring range have as well been fully retained as the typical angular and energy response (see fig. 1). For the FHZ 600 A

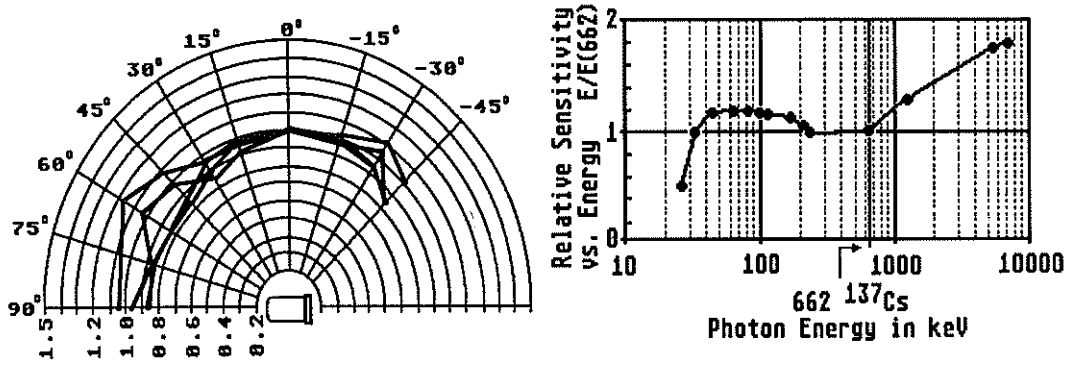


fig. 1: typical angular (48, 65, 118 & 248 keV; normalised to 100% at 0°) and energy response of the FHZ 601 A.

proportional detector as well as for the FHZ 601 A intelligent detector the approval for official calibration by German authority has been given /2/. The individual correlation between a probe and its measuring and display instrument in a calibrated system is no longer necessary. Intelligent probes are calibrated as an independent system. Each of them can be substituted by any other one without loosing the calibration of the system.

2. Electronics

Modern technologies such as SMD technics, flexible printed circuits, and ASIC devices were used to build up the intelligent probe. It contains the energy filtered proportional counter, the high voltage supply, preamplifier, the whole counting electronics and an integrated microprocessor with a serial interface RS 485. The microprocessor controls the whole measurement and transmits a calibrated measuring value to the serial interface. Additionally it permanently performs self-tests for all devices and surveys the operating conditions. The results are transmitted via the RS 485 interface as a status information.

The probe does not require any special display unit. For data acquisition and display a standard PC can be used. A software program for data handling in the PC is available (see below).

3. The FHZ 601 A in Networks

The RS 485 interface permits the construction of local measuring networks, in which up to 32 probes can be operated from a central control station in a simple twisted pair bus system. Each probe can individually be adressed (0...31). Data transmission takes place at 2400 or 9600 bit/s via cables which may exceed even a length of 1000 m. Data transmission over distances of 3 km has successfully been tested /3/.

The measurement information is transmitted by a level 2 protocol (ISO/OSI) listed in DIN 66348 to protect data against transmission errors. The data telegram contains not only the measured value but also an identification parameter and various status information. This telegram permits

- the identification of the source i.e. the transmitting probe
- a check of the data transfer
- a permanent control of the operating conditions and the status.

The permanent self-tests detect any operating conditions out of the normal range and report them as a status information. These tests include:

- measurement of the supply voltage for the digital electronics
- measurement of the temperature
- control of the detector amplification
- preamplifier tests by permanently injecting sets of test pulses
- control of the minimum countrate (detector failure)
- watchdog routines
- test channel for continuously surveying the operating point

Further the calibration and measuring parameters can be read out by the user and a plateau measuring routine is available that can be operated on request.

A built-in history memory stores up to 256 measuring values including time and status. The measurement is therefore independent of the connection to a central computer. On request the history data can be read out without disturbing the actual measurement.

4. Calibration

All the calibration parameters (including two correction factors and the intrinsic background) and related information such as serial number, detector type, and software version are stored in the permanent memory (EEPROM) of each probe. The probe is calibrated at three dose rate levels. This calibration takes place once before delivery. After this initial calibration the calibration data are locked and can not accidentally be changed. For recalibration a hardware switch inside the probe has to be manipulated. Therefore by sealing the detector housing the calibration data are protected against any unauthorized changes.

5. The Intelligent Probe's Family

The electronics of the FHZ 601 A is also integrated into other detectors.

The probe FHZ 621 A covers the middle and high dose rate range from 50 nSv/h to 25 mSv/h. As well as the low range FHZ 601 A the FHZ 621 A is also PTB approved. For the high dose rate range up to 1 Sv/h the FHZ 631 A has been developed.

An intelligent ionization chamber FHZ 191 N covers a wide range from 10 nSv/h up to 10 Sv/h. It has been tested even for application in pulsed radiation fields. Its energy response is nearly constant up to 7 MeV.

The BIOREM FHT 750 neutron detector which uses a BF_3 proportional detector has been combined with the intelligent electronics and thus represents an intelligent neutron detector. This developement allows the integration of neutron and gamma detectors in the same local network.

All these intelligent detectors have the same interface and the same data protocol. They can therefore be used in one network together with any other detectors of this family.

6. MRP Monitoring Radiation Program

As a powerfull tool for analysing and interpreting the measured values a special PC-program has been designed. It allows the acquisition, storage and presentation of the measurement results from up to eight stand alone radiation monitors e.g. aerosol monitors or from up to 6 radiation monitors plus 32 dose rate probes such as e.g. the FHZ 601 A (fig. 2).

According to their interactive communication with the user 3 groups of routines can be found:

6.1. Automatically running routines without interactive control.

These routines are automatically executed after program start up:

- The measuring values of the connected monitors and detectors are periodically acquired. Depending on preset parameters the chart of the last 1 to 31 days of the measured values is graphically displayed. Additionally another set of measured values may be displayed numerically. Fast changing values may also appear in an analoge display as a bargraph.

- Error messages from the remote monitors and exceeded alarm levels as well as transmission failures are signalled optically and acoustically.

6.2. Routines requiring user dialogue

Using the function keys explained on the monitor the following operations may be executed:

- Change of the time and measured value scale of the graphic display.
- Selection of the measured values to be displayed.
- Display of measured values from any time period during the last 10 months and calculation of the average values from a selected interval.
- Numerical output of a complete data set for any selected time.
- Display of a protocol file containing error messages from the remote monitors and excceeded alarm levels as well as transmission failures together with the time they occured.
- Execution of several special functions.

All these functions do not affect the permanent data acquisition from the remote devices.

6.3. Configuration Files

The preset values for the complete program (such as time period of the data acquisition, number of the monitors and detectors, scale and colours of the graphic display and many more) are stored in external ASCII files. This allows the user to modify these parameters and thus to adapt the program to new hardware configurations or to modified requirements.

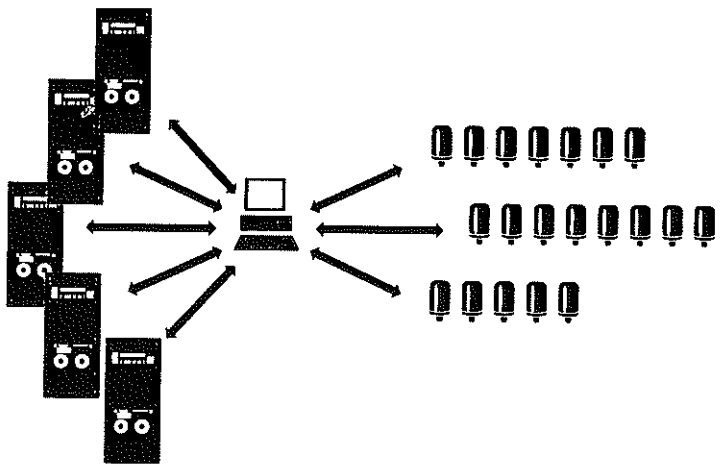


fig. 2:

Local network consisting of aerosol monitors and intelligent probes linked to a PC via MRP

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/2/ Innerstaatliche Bauartzulassungen: PTB Nr. 6.41-06/89 B and PTB Nr. 6.41-17/90 B

/3/ B. Hoffmann et al.; Intelligente Sonde FHZ 601 A für die Ortsdosimetrie; Jubiläumstagung des Fachverbandes für Strahlenschutz e.V.; Aachen (1991)

REPRESENTATIONS OF RADIOECOLOGICAL EFFECTS WITHIN THE *"INTEGRATED MEASUREMENT AND INFORMATION SYSTEM FOR THE SURVEILLANCE OF ENVIRONMENTAL RADIOACTIVITY - IMIS"*

B.W. Bauer, D. Noßke, A. Bayer
Bundesamt für Strahlenschutz, Institut für Strahlenhygiene,
Ingolstädter Landstr. 1, D-8042 Neuherberg/München

Summary

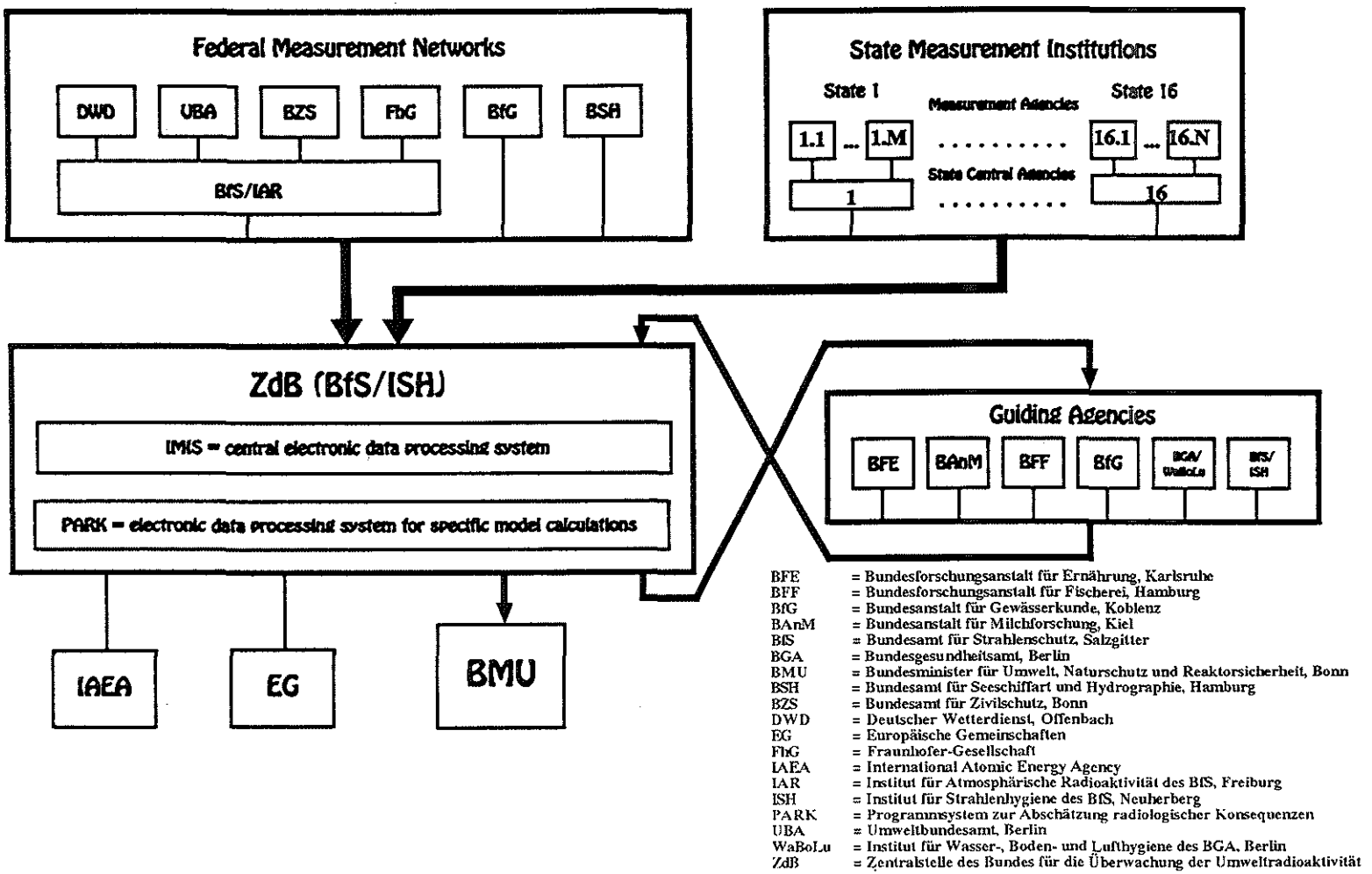
The "Integrated Measurement and Information System for the Surveillance of Environmental Radioactivity - IMIS" pools the measured radioactivity data which come from nation-wide measurement networks as well as from about 50 laboratories. These data are processed and uniformly presented in certain types of diagrams, e.g. maps for geographical overviews and curves for time dependencies. During the course of an "event with possible non-negligible radiological consequences" the IMIS will provide at any time comprehensive information about the radioecological situation, represented in an easily interpretable form. This supplies the basis for decisions about preventive measures.

IMIS: Organization and Data Flows

In order to carry out the tasks laid down under the "Precautionary Radiological Protection Act" the "Integrated Measurement and Information System for the Surveillance of Environmental Radioactivity - IMIS" is currently being developed. These tasks are the continuous monitoring of environmental radioactivity and, with the same significance, the minimizing of the radiation exposure of people in the case of an "event with possible non-negligible radiological consequences" [1,2,3]. IMIS is a decentralized system of computers which are linked to the Central Federal Agency (ZdB); the installed software consists partly of customary products and partly of especially developed programs.

The organization scheme with the most important institutions involved and the main data flows is shown in fig. 1. The environmental radioactivity data are ascertained by Federal Measurement Networks as far as they concern the large-scale contamination of air, ground surface, surface waters, and by State Measurement Agencies as far as they concern individual materials like foodstuffs, animal feed, soil and plants, drinking water, and waste. These data are transmitted to the ZdB within adequate short time intervals. The Guiding Agencies check specific data sets for consistency with other relevant information.

The real-time program system PARK, "Programmsystem zur Abschätzung radiologischer Konsequenzen" [4], which is developed at the GSF (Forschungszentrum für Umwelt und Gesundheit, Neuherberg/München), assesses the radioecological effects in the form of specific activities of e.g. foodstuffs as long as measurement data are not yet available, and in the form of different dose values resulting from given radioactive contamination.



BFE = Bundesforschungsanstalt für Ernährung, Karlsruhe
BFF = Bundesforschungsanstalt für Fischerei, Hamburg
BfG = Bundesanstalt für Gewässerkunde, Koblenz
BAnM = Bundesanstalt für Milchforschung, Kiel
BfS = Bundesamt für Strahlenschutz, Salzgitter
BGA = Bundesgesundheitsamt, Berlin
BMU = Bundesminister für Umwelt, Naturschutz und Reaktorsicherheit, Bonn
BSH = Bundesamt für Seeschiffart und Hydrographie, Hamburg
BZS = Bundesamt für Zivilschutz, Bonn
DWD = Deutscher Wetterdienst, Offenbach
EG = Europäische Gemeinschaften
FhG = Fraunhofer-Gesellschaft
IAEA = International Atomic Energy Agency
IAR = Institut für Atmosphärische Radioaktivität des BfS, Freiburg
ISH = Institut für Strahlenhygiene des BfS, Neuherberg
PARK = Programmsystem zur Abschätzung radiologischer Konsequenzen
UBA = Umweltbundesamt, Berlin
WaBoLu = Institut für Wasser-, Boden- und Lufthygiene des BGA, Berlin
ZdB = Zentralstelle des Bundes für die Überwachung der Umweltradioaktivität

Fig. 1: IMIS: Organization scheme and main data flows

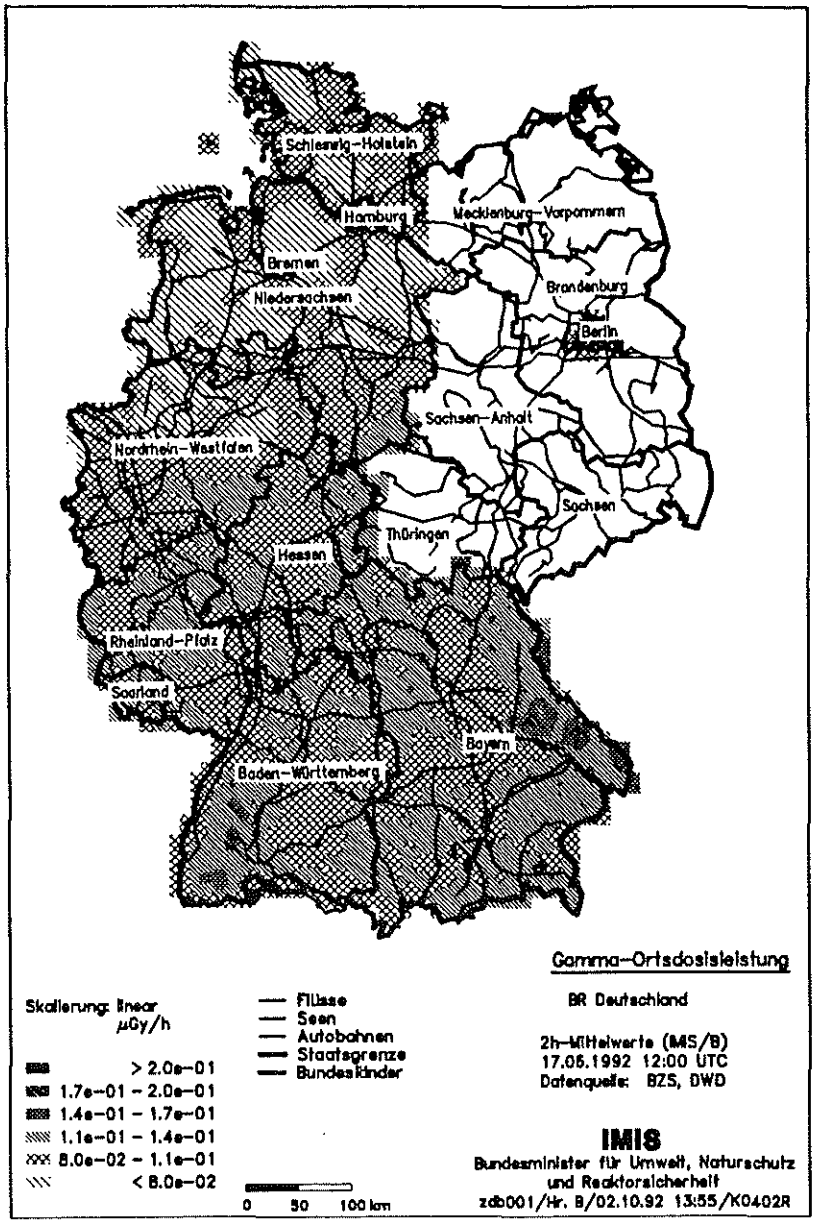


Figure 2

Types of Data Representation

Within IMIS the registered and evaluated data are represented in several main forms: maps, curves, tables, and bar charts. In principle, all data can be represented in all of the different types of diagrams.

Maps provide general overviews over the radioecological situation in Germany; fig. 2 shows as an example the gamma dose-rate for 17 June 1992 (the original figure is usually coloured). Since the measurement network in the former GDR is still being built up, the map contains no measurement information in that area.

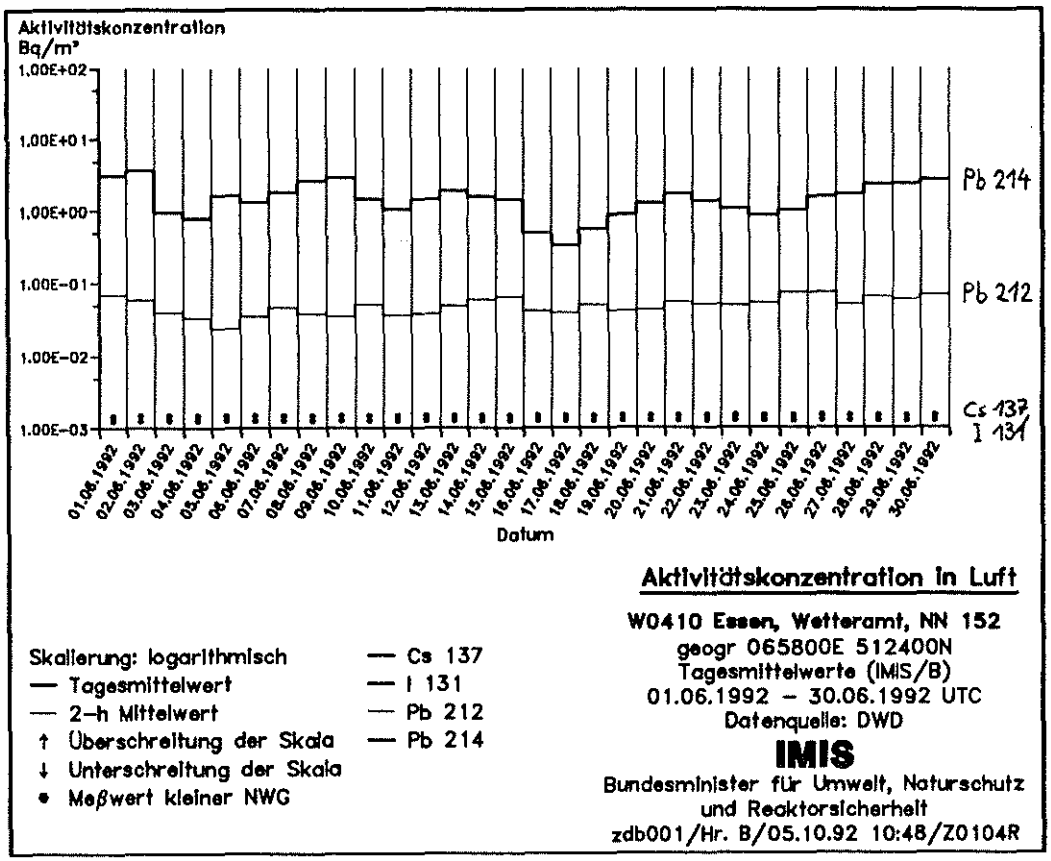


Figure 3

In order to judge temporal trends a chronological sequence of maps is required. For a selected individual measurement station the local time course is presented as histogram; fig. 3 (originally coloured) shows the measured specific activities of single nuclides in air at the 'Wetteramt Essen' during June, 1992. Since the activities of relevant nuclides do not exceed the detection limits, the values for lead isotopes - Radon daughter products - are shown.

After the early period of a radioactive release event, the contamination of e.g. plants and foodstuffs becomes increasingly important as the source of doses by ingestion. The related data would be presented in regional maps; the smallest representable units are the "Landkreise". For comparisons with e.g. existing limiting values, the numerical presentation in tables is required, usually given for individually selected parameter sets of environmental section, region, radionuclides, and period.

For answering particular questions like the contributions of different contaminated foodstuffs to the ingestion dose bar charts are useful tools. Data of this type result from PARK; the especially adapted representation forms are currently being developed.

IMIS-Application during an Emergency Event

During a situation of a potentially large-scale radioactive contamination of the atmosphere the proceeding is directed on three main topics (for a more detailed discussion cf. [5]).

Primarily spatial overviews over the measured data are needed in form of maps. It is then easily possible to distinguish between regions without and with significant radioactive contamination and to focus on the latter. Temporal trends can be analyzed by a sequence of maps or by curves for selected individual measurement stations.

When a critical region is identified, in a second step the interest concentrates on the contamination of various environmental sections due to different radionuclides; the corresponding specific activity data which have been estimated by PARK or already measured are visible in form of maps for particular combinations of environmental section and nuclide or tables for certain places and environmental sections. The data can be compared with limiting values or intervention levels, as far as they exist.

The final effects for the population are quantified in the form of different dose values assessed by PARK on the basis of given radioactive contamination data. Such data comprise a variety of constellations of place, exposure pathway, environmental section - especially foodstuff for ingestion -, radionuclide, organ, age class. These informations help to influence the consequences of an emergency event via adequate precautionary measures.

Concluding Remarks

While some types of questions - e.g. nation-wide overviews over the radioactivity in air - can easily and quickly be answered by the existing means, certain other types of questions - e.g. a spatial representation of the development in time - require (at the moment) combinations of different types of data representations, sometimes also further information from the outside of IMIS. Nevertheless, the explained graphical representations of information about environmental radioactivity are useful tools for a quick and reliable assessment of the radioecological situation and supply the basis for decisions about precautionary measures.

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THE *SYTAR* NETWORK: A RADIOACTIVITY OBSERVATORY

Daniel ROBEAU et Dominique CALMET

Institut de Protection et de Sûreté Nucléaire
Département de Protection de l'Environnement et des Installations
BP 6, CEA-FaR 92265 Fontenay aux Roses CEDEX

Summary:

A Radiological Transmission and Early Warning System (SYTAR) connects a network of stations all over France to the Emergency Centre of the Institut de Protection et de Sûreté Nucléaire in Fontenay aux Roses. Under normal circumstances, radiological data collection is done in three different ways, (i) a real time transmission of total alpha and beta in aerosols and external gamma radiation measurements performed and relayed by beacons fitted with sequential filters; (ii) a transmission of recorded data containing results of alpha spectrometry: weekly beta and gamma measurements of aerosols and rainwater and monthly surface deposition and measurements of biological indicators, (iii) in accident conditions the rate of data delivery is stepped up.

1. Introduction

Starting in 1953, atmospheric fallout from nuclear weapons tests in the northern hemisphere prompted the setting up of a monitoring network to measure radioactivity in air. Together with the development of the nuclear industry, networks to monitor radioactivity in various segments of the environment were subsequently set up. In 1989, a Radiological Transmission and Early Warning System (**SY**stème de **T**ransmission et d'**A**larme **R**adiologique, acronym: **SYTAR**) came into being in order to harmonize radioactivity surveillance methodologies and furthermore to trigger off an alert throughout the networks whenever an unusual degree of radioactivity is detected. **SYTAR** is a remote permanent system linking up a national electronic access and the radiation protection services of nuclear facilities as well as selected reference sites located on the mainland.

The radioactivity measurements done in the framework of the SYTAR network are designed to provide consistent radiological coverage of all nuclear facilities and hence contribute to the monitoring of the national territory. This network is structured according to three alarm levels:
- real time monitoring of total atmospheric radioactivity;
- deferred time spectrometry of air and water samples as well as ground deposition, and
- monitoring of the significant link in the food chain, also in deferred time.

The equipment and the process to determine levels of radioactivity are geared to radiological protection recommendations. A rapid data transmission system from monitoring sites to the Emergency Centre of the Institut de Protection et de Sureté Nucléaire (IPSN), pools all incoming measurements in order to provide the relevant authorities with all the information required to assess the radiological impact of an abnormal event occurring within or outside France.

2. SYTAR Modalities of Operation

All CEA Group facilities have a Radiological Protection and Environmental Surveillance Service in charge of the radiological monitoring of the environment surrounding the facilities, in order to:
- check the smooth operation of the installations and verify that low level radioactive discharges are within authorized limits;
- take immediate steps to determine the impact of possible accidental discharges in the event of an accident or incident involving on-site devices or equipment as well as assist relevant authorities, if required, to take action in order to protect staff and nearby populations and furthermore curtail contamination of the facilities and the surrounding environment.
- ensure surveillance of radioactive contamination, from whatever source, affecting a substantial portion of mainland territory. To this effect, a co-ordination network was established between the existing surveillance teams, drawing on the means of the various CEA Group sites, then supplementing them and making sure that information obtained is made use of and circulated.

The network is also used during normal operation to provide the authorities with information on the environmental level of radioactivity at the CEA Group nuclear facilities. Above and beyond monitoring the environment, this network has three further objectives in the event of an accident:
- detecting the appearance of an abnormal radiological situation due to a domestic or foreign source of contamination and triggering the alert;
- monitoring the evolution, determining the components and forecasting the public health implications of any discharges and determining monitoring and remedial action as necessary in case of severe impact;
- determining the surveillance and counter-measures necessary if the scope of the consequences so justify.

To satisfy these 3 objectives, SYTAR operates at three levels:
- **Monitoring level 1** corresponds to **global, permanent** radioactivity measurements of air. Results are obtained in real time and transmitted to a Central Post at each nuclear centre. Each nuclear facility to which the real time results are transmitted is capable of triggering the alarm.
- **Monitoring level 2** is implemented once the alarm is set off. It supplements the level 1 measurements with measurements of specific, deferred radioactivity concerning the physical environment.
- **Monitoring level 3**, initiated on the basis of results obtained during monitoring level 2, consists of, in addition to the continuation of level 1 and specific measurements of radioactivity, made on biological samples of the food chain.

The following elements have been chosen to determine contamination or exposure values that will trigger the alarm and when to change over to defined monitoring:
- for measurements of external gamma irradiation and the corresponding activity of gases and aerosols, significant variations in ambient radioactivity;
- as regard the physical environment, values calculated from tolerable levels for food stuff and the recommendations of the International Commission on Radiological Protection (ICRP).

3. The SYTAR Network

The general structure of the SYTAR network is based on sites that are able to perform three types of surveillance. Surveillance may differ according to equipment and available staff.
Type A sites participate in monitoring at level 1, and set of the alert. They also participate in monitoring at levels 2 and 3.
Type B sites participate in monitoring at levels 2 and 3.
Type C sites participate in monitoring at level 1.

All sites are linked to the IPSN. Information is received, analysed and disseminated at all times by the IPSN, enabling it to take the required decisions. The technical centre is installed in the Fontenay aux Roses Research Centre and is operated by the Département de Protection de l'Environnement et des Installations.

3.1. Type A Monitoring Sites

The Centres at Saclay, Fontenay aux Roses, Cadarache, Grenoble, Marcoule, La Hague, Monthlery, Bruyères-le-Chatel and Valduc participate in level 1 monitoring, giving the Network Alert and in monitoring at levels 2 and 3. They are capable of indicating any abnormal situation and providing validated information which, if necessary will be the basis for setting off an alarm. Information validation consists both of validation of the measurement itself and of the result obtained, and an attempt to localize the source of release be it outside or inside the relevant site. For sites from which there are radioactive discharges into the atmosphere, the means habitually used to detect this source, is to have several monitoring stations and a meteorological station.

The parameters monitored permanently (level 1) to trigger the alarm are:
- Global alpha and beta radioactivity of aerosols (detection limit less than 5.10^{-2} Bq.m^{-3});
- external irradiation due to gamma radiation (detection limit less than 10 nGy.h^{-1});
- meteorological parameters, at least wind speed and direction, thermal gradient and rainfall.

The physical environment is also monitored (level 2). This involves obtaining and later analysing samples in order to identify the main radionuclides present and determine their radioactivity. The parameters are:
- radioactivity of aerosols
- radioactivity present in the soils (total deposit)
- radioactivity of surface and underground waters and marine waters.

The action limits indicated in table 3.1.a were derived from assumptions made concerning radionuclide transfer to food stuff. These values are distinct from the detection limits indicated previously, which correspond to the characteristics of the measuring instruments.

Indicators	Action limits	
Aerosols	5.10^{-1} Bq.m^{-3}	(Beta et gamma emitters)
	1.10^{-1} Bq.m^{-3}	(Alpha emitters)
Vapours	5.10^{1} Bq.m^{-3}	(Iode-131)
	1.10^{2} Bq.m^{-3}	(Iode-133)
Gas	1.10^{3} Bq.m^{-3}	(Gamma emitters)
Total deposit	5.10^{2} Bq.m^{-3}	(Beta et gamma emitters)
	1.10^{2} Bq.m^{-3}	(Alpha emitters)
Surface water	8.10^{1} Bq.l^{-1}	(Beta et gamma emitters)
	2.10^{0} Bq.l^{-1}	(Alpha emitters)

Table 3.1.a: Action limit for physical parameters.

At level 3 the monitoring network comprises measurements along the food chain of samples of sensitive products that are characteristic of the region and are already routinely measured according to the modalities defined. These are analyzed in order to identify the main radionuclides present and determine their radioactivity.

The products to be monitored are:
- vegetable produce:

  Food for human consumption such as vegetables and fruit;

  Animal fodder such as meadow grass.
- Elements of the human diet such as milk and drinking water.

Sampling methodologies are established for each category of product to be monitored. The action limits are specified in Table 3.1.b.

Radionuclides	Sensitivity for		Value reached for		Time required
	milk Bq.l^{-1}	other food Bq.kg^{-1}	milk Bq.l^{-1}	other food Bq.kg^{-1}	hour
^{131}I	50	200	0.5	2	1
^{90}Sr	10	75	2	2	75
^{137}Cs and ^{134}Cs	100	125	1	2	2
^{239}Pu and ^{241}Am	2	8	0.1	0.1	72

Table 3.1.b: Action limit for food.

3.2. Type B Monitoring Sites:

They are not taken into account for the triggering of the network, but their data are used during the emergency situation.

They complement the type A sites. Any site able to take some or all the samples indicated above, according to the defined operation procedures, and external irradiation measurements, can be integrated into the network. It will also ensure the fast transmission of its results to the IPSN. The type B monitoring nodes are IPSN Stations of Orsay at level 2 and 3, Verdun, Le Barp, La Seyne, Flers, Papeete and La Réunion at level 2.

3.3. Type C Monitoring Sites

These sites participate only in alerting the network, but do not participate at monitoring levels 2 and 3 during the entire duration of the emergency situation. The type C monitoring nodes are EdF nuclear power plants in Fesseinheim, Cattenom, Gravelines and Le Blayais.

4. Operation of the Network

4.1. Monitoring Level 1

Stringent operation modes are designed for measuring instruments qualified for use in the network. Results are transmitted in real time to IPSN in Fontenay aux Roses where they are processed and filed. In order to train the staff and check the good operation of these transmissions, results are validated and transmitted on a weekly basis via acquisition menus:

- an acquisition menu which is transmitted periodically and, in addition to the site references, mean minimum and maximum values of the measurements are carried out during the relevant period;
- an acquisition menu containing data relative to the equipment and which need to be constantly updated.

4.2. Monitoring Level 2

Once an emergency situation is confirmed and validated, the first task at monitoring level 2 is the characterisation of the radionuclides in question both as regards quality and quantity. This characterisation is carried out in the physical environment, atmospheric and soil deposition.

It is the IPSN Emergency Centre which decides when sites must change over to level 2 by putting into application measurement procedures aimed at establishing the consistency of the results for their later processing. All A and B sites will change over barring exceptional cases. Sites should proceed to measurements in a synchronized fashion. The emergency centre determines measurement frequencies, which may be 24 hourly, 12-hourly, 6-hourly, or indeed, in extreme situations, 3-hourly. Sampling will be synchronized in time and duration to facilitate the work of global interpretation. Deposit measurements, taken directly or from samples, are recorded indicating the time.

The IPSN Emergency Centre informs the designated authorities and the network sites of the results collected, together with an analysis of public health implications. All data received by the IPSN Emergency Centre may be consulted at any time by the different nodes in the network. The Emergency Centre decides to launch the procedure for change-over to level 3 according to the interpretations concerning public health.

4.3. Monitoring level 3

This level implies a possible modification of level 2 essentially as regards measurement frequencies in the physical environment and change-over to the analysis of the food chain. This means stepping up the frequency of biological samples habitually carried out within the context of site monitoring. All the measurement results are transmitted to the Emergency Centre as before. Health interpretations are made and transmitted to the relevant authorities and to the network sites.

4.4. Accidental situations

The alert is trigger locally when one, or more, predetermined alert limits are exceeded and the excess values have been validated. The procedure for transmission to the IPSN is then undertaken.

4.4.1. Alert Action Limits

For the two primary parameters measured, namely aerosol radioactivity (contamination) and dose rate (irradiation), alert limits are set at

- 0.2 $Bq.m^{-3}$ for global alpha and beta radioactivity of aerosols;
- the five to ten fold site mean value for ambient dose rate.

Since the emergency should not be triggered off by higher controlled discharges but should be set off by a level, albeit low, that would nevertheless be higher than the average value, a compromise was aimed at in setting the values. **The values may be adjusted according to feedback based on experience for each site.**

4.4.2. Validation of Alert Results

When a value exceeding the action limit is detected, the first response is to validate the data before transmission. There are two aspects to this validation: validation of the physical measurement and validation of the radiological data. This is based on several criteria such as:
- simultaneous exceeding of the alert limit at two or more stations,
- kinetics and duration of the phenomenon. There should be a possibility to carry out rapid additional measurements.

As soon as the data is validated, it is transmitted to the IPSN and the site is placed immediately in the situation of monitoring level 2, if equipped to provide this function.

5. Simulated Operation of SYTAR in accident circumstances: Cadarache drill (14-17 October 1991)

Accident conditions are simulated periodically. The Cadarache drill enabled the testing of the SYTAR network under accident conditions. In the course of the drill, the relevance of the organizational flow chart, the methods and means of action implemented in order to characterize extended contaminated areas, transmission methodologies, presentation procedure of results to the competent authorities and operational routes of communication were tested. The performance of mobile teams (teams in the field) and assay laboratories were assessed.

The simulation scenario prepared by the IPSN gave the description of a serious accident that has just occurred at a reactor (partial core melt-down) located in the Cadarache Research Centre. The reactor was releasing an unspecified amount of radionuclides per hour into the atmosphere. The competent authorities (préfet) requests CEA action. In accordance with the scenario the releases ceased in the morning of 15 October and the teams set off to do the sampling. The samples were collected at an outpost area where they were processed and sent on to the measurement laboratories readied for the occasion i.e. the Radiation Protection Service laboratories of Fontenay aux Roses, Cadarache, Grenoble, Marcoule and Pierrelate facilities in addition to regional agriculture laboratories and to the Forestry Laboratory for the Provence Alpes Côte d'Azur regions, located in Marseille.

Samples gathered in the field were not radioactive. Each sample sent on with an envelope containing the presumed radioactivity value had the release actually occurred. Other samples had been previously labelled with ^{137}Cs and ^{131}I so that measuring could actually take place. Each laboratory got twenty-two labelled samples and some twenty non-labelled samples. Samples shipped by car or plane reached the laboratories in mid-afternoon and were then measured. Results were forwarded to the Technical Emergency Centre through SYTAR (Radiological Alert and Transmission System). A major part of measurement findings had been pooled at the Emergency Centre by 9 p.m.

Meanwhile, the outpost transmitted some one hundred gamma irradiation measurement results from radioactive airborne/aerosol particles deposited on the ground. The findings of measurements done at the Cadarache facility were transmitted to the Emergency Centre through the same SYTAR network with a mobile focal point established at the outpost.

Upon resumption of the drill on 16 October, the Emergency Centre was in a position to draft summary findings with charts and maps that were subsequently referred to several authorities. These charts and maps collated the results of measurement work done on substances meant for human consumption or environmental samples.

6. Public information

The French general public has access to radiological data via the Minitel network. Anybody who has a telephone line has access to these services. By linking up to a display screen the telephone user may connect with a number of data banks and services. Users have access to several Minitel services providing information on radioactivity levels in France, particularly for areas surrounding nuclear facilities. Two national services provide similar information: the MAGNUC service managed by the Ministry of Industry and the TELERAY service of the Ministry pf Public Health. The MAGNUC service mainly gets input from SYTAR network measurements. The service provides radioactivity data for each SYTAR network facility as well as for EdF facilities that are not part of the network. The MAGNUC service provides a monthly average radioactivity level for a ten km area surrounding each nuclear facility.

THE NETWORK FOR AUTOMATIC DOSE-RATE ALARMING AND MONITORING *"NADAM"*

). Rauber, D. Sulmoni Thomi
lational Emergency Operations Centre, Postal Box, CH-8044 Zürich

ıbstract

:he network for automatic dose-rate alarming and monitoring NADAM :onsists today of 58 stations, distributed at different locations)f the meteorological network ANETZ all over Switzerland. The .ocal dose-rate is continuously measured with a Geiger-Müller-:ounter. The measured value is transmitted to a central computer ıt the Swiss Meteorological Institute (SMA) every 10 minutes. If .t exceeds a threshold of 1 μSv/hr an alarm is given. The data are .nterpreted by means of meteorological parameters and by :omparison with neighbouring stations. Simultaneously measurements ;ith other instruments are used to obtain further informations for :he interpretation of the results. A part of these data is weekly)ublished in the Bulletin of the Swiss Federal Office of Health ınd daily in TELETEXT (Swiss TV).

Iistory and actual state

:n the late Seventies the need for a network for automatic and :ontinuous monitoring of local dose-rates was growing.
ı project was initiated in 1977. One of the most important :haracteristics of the network was to position the detectors at)laces where also meteorological stations were available. ıdvantages of this solution were that indications of possible :orrelations between local dose-rates and meteorological)arameters became available and that the recently installed ıetwork ANETZ of the Swiss Meteorological Institute could be used :or transmission and first processing of the data.

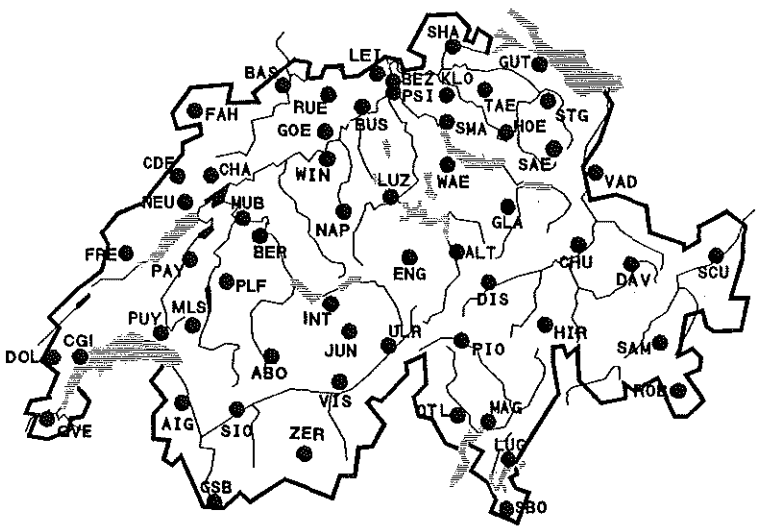

Figure 1:

The 58 stations of the NADAM network in 1992.

Gloor Inc. constructed the detectors using GM counters. The development of an optimal detector was a difficult task: it had to resist to hard environmental conditions, to work in a relatively large range (50 nSv/hr to 50 mSv/hr) and to be integrated into ANETZ [1].

The NADAM network started in June 1982. In this pilot phase 8 of the then planned 51 stations were in operation. In 1986 when the Chernobyl cloud reached Switzerland, there were 12 stations. Following this event the network was quickly finished. Today there are 58 NADAM stations. The distribution of the stations reflects the population density, but there are also high-altitude stations and five special stations at the Swiss nuclear power plants and at the Paul Scherrer Institute (PSI); an additional station was set up at Vaduz, Liechtenstein (Fig. 1).

Data processing

The continuously measured data are transmitted in 10-minute intervals (as 10-minute averages) to the central station at the Swiss Meteorological Institute in Zurich which automatically checks every value and triggers an alarm if the alarm level of 1 μSv/hr is exceeded.

After this first check the data are stored in a data base and processed to get hourly, daily and monthly averages. Daily plausibility checks allow to detect faulty detectors. Many deviations from normal behaviour can be explained using the meteorological parameters.

Publications

Every day and for 16 selected stations (representative regions and all nuclear power stations) the daily averages are published in

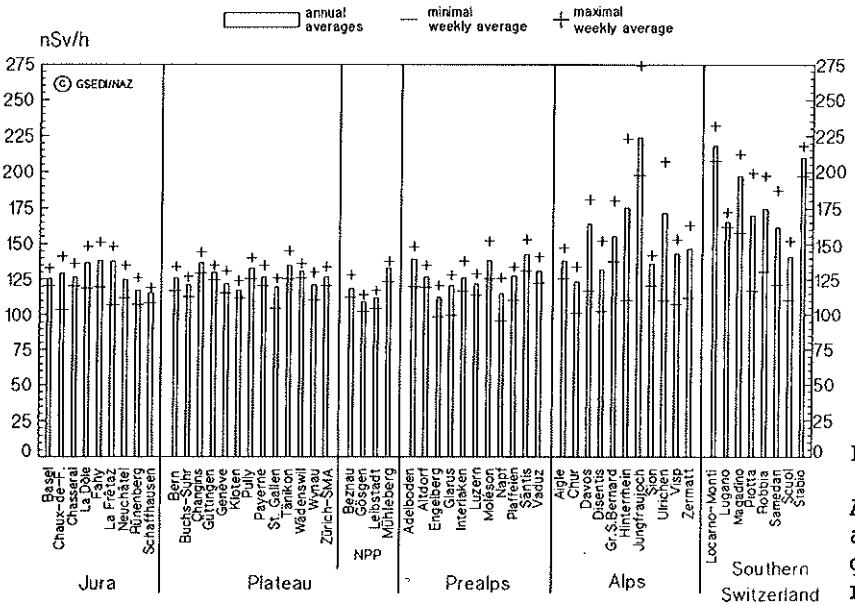

Figure 2: Annual averages 1991 grouped by regions.

text TV TELETEXT.

Every week the Bulletin of the Federal Office of Health (BAG) contains the weekly averages of all stations, extreme values with explanations, and graphics of the daily averages during the past month with maximal and minimal 10-minute values for the 16 selected stations. These graphics also show the precipitations.

Once a year a review of the situation of the network and of the results of all NADAM stations is published in this Bulletin and in the Annual Report on Radioactivity of the Federal Office of Health.Figure 2, taken from last year's report [3], shows the annual averages and the maximal and minimal weekly averages of all NADAM stations grouped by regions.

Some results

From the beginning the correlation between local dose-rate and meteorological parameters was recognised [1, 2]. Especially the effect of increased dose-rate during intense precipitations with wash-out and deposition on the ground of atmospheric radioactivity is impressive. Figure 3 shows such an event. A sudden rain shower in Vaduz washes out the atmospheric radioactivity. The local dose-rate increases considerably (200% of the normal value) and then decreases exponentially. The half-life is about 2 hours.

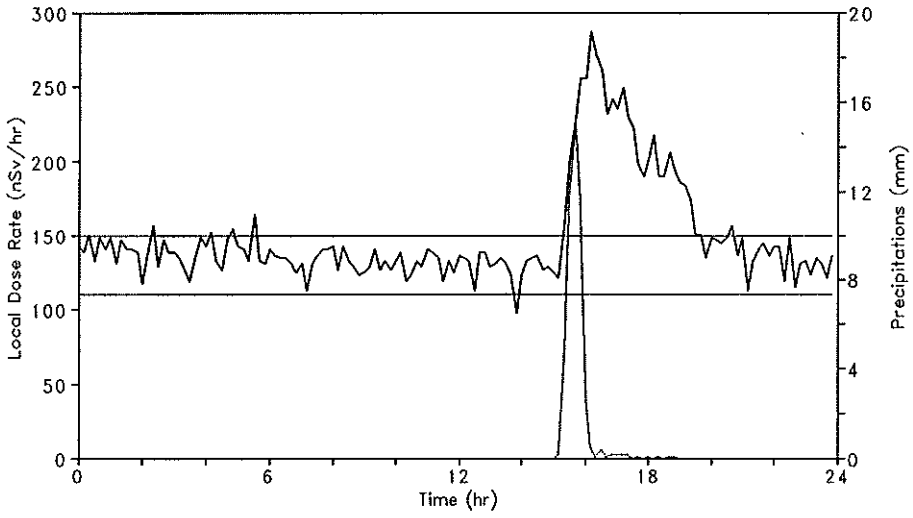

Figure 3: Local dose-rate (10-minute averages) and precipitation rate at VADUZ (24.06.1991).

Other correlations are deducted from comparing neighbouring stations. Figure 4 shows the weekly averages of the local dose-rates of the stations Aigle, Interlaken, Visp and Adelboden. The curves show clear similarities. Using such graphics faulty detectors can be identified. During a contamination such informations may help to find possible differences of the contamination situations.

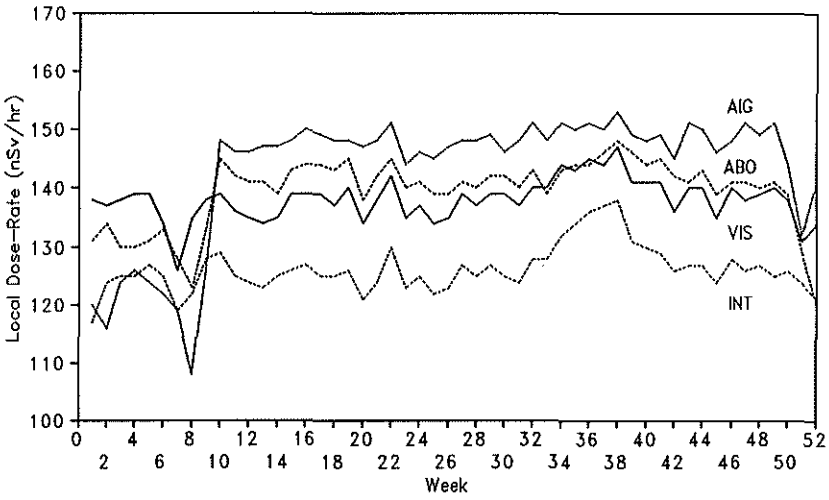

Figure 4: Annual variations of local dose-rates 1991 at four stations.

During winter the curves seem to allow less comparison. This depends on the snow cover which acts as an absorber; the detectors thus measure a reduced ground radiation. This effect is shown in Figure 5 for the station Hinterrhein. Melting of the snow causes a continuous increase of the local dose rate.

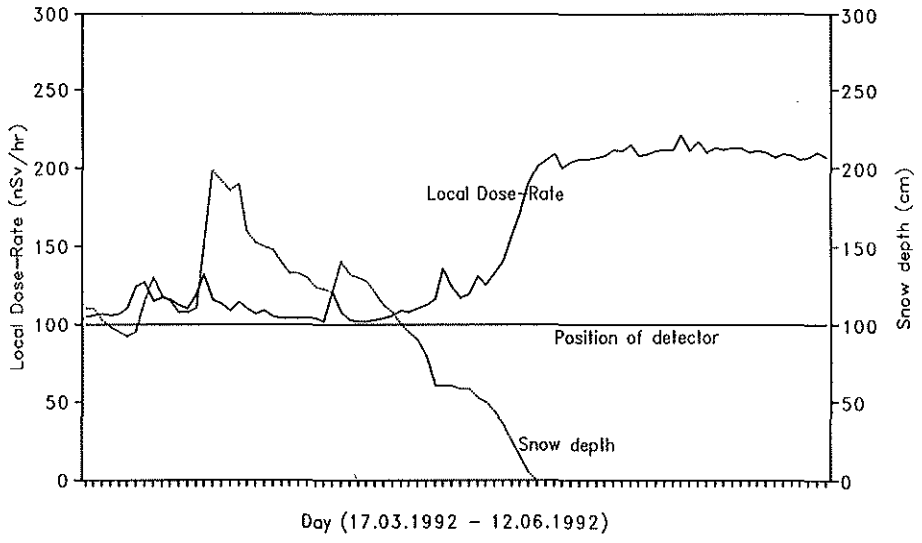

Figure 5: Effect of snow depth on local dose-rate (detector located at 1 m above ground)

omparison with other instruments

or comparison and control each NADAM station contains a hermoluminescent dosimeter (TLD) which is read quarterly. During he annual maintenance the service personnel performs parallel easurements with portable dose-rate monitors Automess 6150 AD 2.

ll these comparisons have shown that the NADAM detectors produce ather too high values of local dose-rate at natural background evels.

he reasons for these differences of the NADAM detectors are resently investigated by the Radiation Hygiene division of the aul Scherrer Institute. Comparisons with other instruments under ifferent measuring conditions allow to exclude incorrect alibration or geometry effects as causes for the differences. The ffect rather seems to consist of a background and of a different esponse to cosmic radiation (energy dependence at high energies) 4].

espite these problems the NADAM detectors fulfill their alarm unction and can be used as reliable monitors in case of ground ontaminations, their use for monitoring normal background is only secondary task.

eferences

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THE SWISS *MADUK-ANPA* PROJECT FOR DOSE RATE MONITORING AROUND THE NPP's *(MADUK)* AND FOR EMERGENCY RESPONSE DATA TRANSFER *(ANPA)*

Franz Cartier, Peter Uboldi
Swiss Federal Nuclear Safety Inspectorate, HSK
CH-5232 Villigen-HSK, Switzerland

Summary

Switzerland is implementing a dose rate remote monitoring network, called MADUK, with from 12 to 17 sensors placed around each of the four NPP-sites at distances up to 5 km.

In addition, an automatic data transfer system, called ANPA, will be installed, which will transmit the most vital parameters of the reactor facility to the Swiss Federal Nuclear Safety Inspectorate (HSK) during an accident.

It is planned to realize the integrated MADUK-ANPA system by the end of 1993.

1. Introduction

The HSK has the duty to supervise and to verify that the licensee takes all the necessary actions for the protection of the population and the environment. In emergency situations, the HSK also supports the Swiss Emergency Center (NAZ) in the assessment of the possible radiological consequences outside the plant.

To cope with this task, the HSK maintains an emergency organisation with a person on standby 24 hours a day. With the system discussed here, the HSK emergency organisation will be supported by a reliable and swift information transfer directly from the power plant and dose rate monitoring network to its emergency facility in Würenlingen.

2. Overview of the MADUK-ANPA Project

Figure 1 shows schematically the organisations involved in this network and the information flow established between various data groups. The system constellation reflects the existing organisational and technical conditions.
Data sources are the process computers of the NPPs at Leibstadt, Beznau, Gösgen, Mühleberg (named KKL, KKB, KKG, KKM), the gamma dose rate monitoring system and the meteorological instrumentation in their vicinity.
Data transmission will be accomplished over a X.25 wide area network based on lines leased from the Swiss National Telephone Company (PTT). The data from the meteorological tower at each NPP site are regularly transmitted to the Swiss Meteorological Institute (SMA) since the startup of the NPPs.
The Data Center is located at the HSK offices in Würenlingen. There, the periodically requested dose rate from the monitoring network, meteo data from the SMA and data from the plant will be taken over by the server of the hierarchically assembled system. The data are checked, processed, stored and presented to the user in the requested form.

Data users will be the HSK for all available data, the Swiss Emergency Center (NAZ) in Zürich, the Section SUER of the Federal Office of Public Health in Fribourg and the NPPs for the radiological data only (emission data and gamma dose rate). The data access is controlled by user dependent privileges.

3. Emergency Response Data Transfer (ANPA)

The plant parameters to be transmitted during an accident are of use for the assessment of the following aspects:

- The actual condition of the core cooling and the systems involved with core cooling and decay heat removal
- the state of the barriers that enclose radioactive materials, especially the containment
- the state of the systems that retain radioactive materials (such as filtering facilities)
- the emission of radioactive materials, as well as
- the forecast of any possible forthcoming increase of radioactivity.

The selection of the ANPA parameters is based on the German KTA regulation 3502 "Instrumentation for Emergency Conditions".

The transfer of ANPA data by the NPP will be accomplished by means of a front end computer in such a way, that there is no possibility of influencing the power plant in any way from the outside. Data transmission is only started after mutual agreement. The data transmission is explicitly not intended for surveillance of the NPPs by the HSK during normal operation.

The plant data are transmitted every 2 minutes and are stored for 5 hours. The emission data are 10-minute values (like the meteo-data) and are stored for 72 hours.

For fast detection and tracking of an emergency, the plant parameters will be displayed in static and dynamic graphics (Figure 2) for the following parts of the facility: reactor coolant system, secondary system, emergency cooling and decay heat removal system, containment and radiation monitoring system. For additional analysis of the plant parameters, further programms will be developped.

4. Dose Rate Monitoring System (MADUK)

MADUK has to accomplish the following tasks:

- During normal operation the sensor network must monitor the environmental gamma dose rate
- during operational disturbancies local radiation dose rate increase has to be monitored
- during an accident or severe accident it should give an indication of the actual radiological situation, confirming the endangered sector and the areas not affected.

The location of the dose rate monitors were determined according to the following criteria (in order of descending priority):

- at least one monitor in each political community in zone 1 (radius up to 5 km)
- main impact points according to the meteo statistics of the region
- population centers/densities around the emission site
- one monitor in each 30°-Sector, if possible at a distance of about 1 km.

As a rule, the gamma dose rate sensors have to be installed in the open air , one meter above cultivated ground (grass, lawn) near (mostly public) buildings, where the additional electronics for the monitors can be located. The distance of detectors from the nearest building has to be at least equal to the heigth of the building. In a few cases the sensors must be placed on flat roofs.

The dose rate monitors are equipped with two Geiger-Müller-counters to measure the dose rate range from natural background radiation up to several Sv/h.

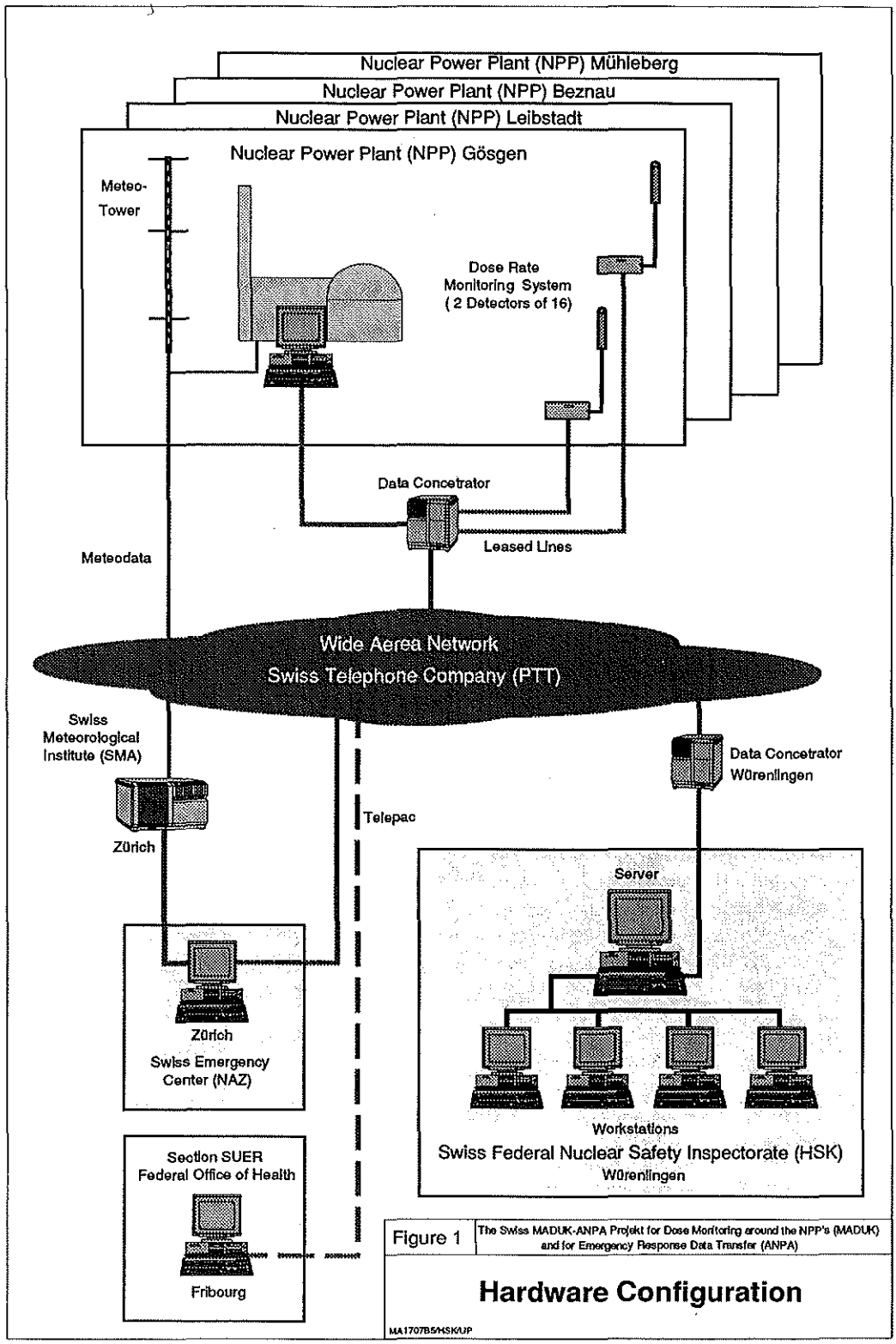
The dose rate values generated and transfered in 10 minutes intervals will be analysed on line in the data center. Net dose rates will be calculated by correlation analysis between the points of one NPP site, so that any incident involving radioactive emissions or radioactive deposits may be recognised at an early stage. The net dose rate exceeding preset values will trip a warning to the responsable person at the HSK, who then will undertake the necessary actions.

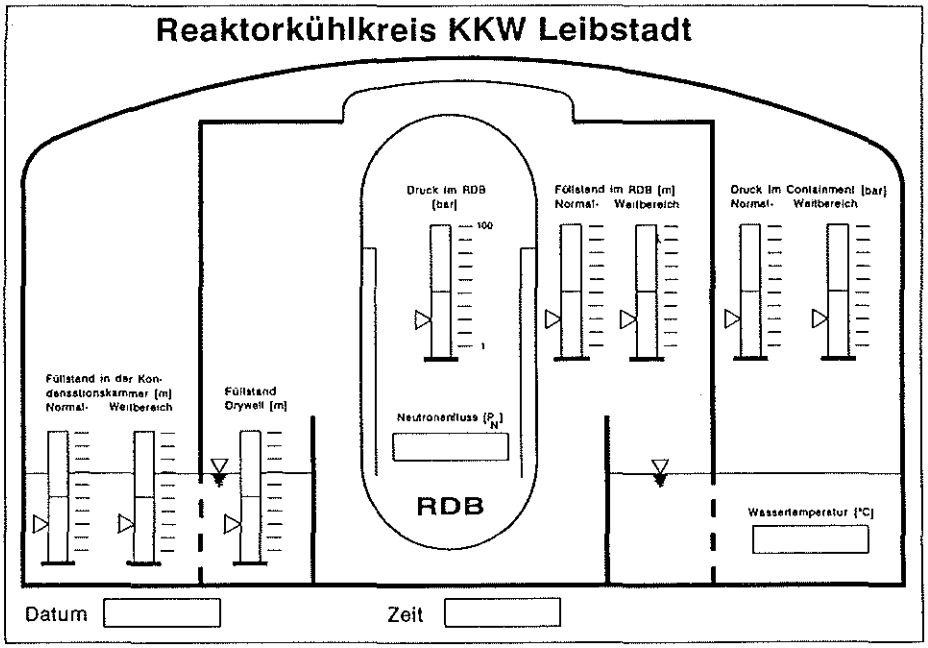
Further, a software program will be installed to estimate the radiation exposure of persons in the vicinity of a NPP based on actual or potential NPP emissions and meteo-data:
- in normal operation and during incidents this allows a comparison with measured dose rates (gamma submersion and radioactive deposition)
- during an emergency situation it allows a fast, preliminary, rough estimation of the exposures in zone 1 and 2 (up to 20 km).

Principally, dose calculation is accomplished in two stages:
- Calculation of the atmospheric dispersion by a quasi-time-dependent Gaussian dispersion model based on the meteorological data, considering plume rise, orography, washout and fallout. A polar-coordinated grid with a 5 degree resolution up to a distance of 20 km will be used.
- The potential dose will be calculated according to the dose model of the general administrative regulation of Germany (§ 45 of the radiation protection regulation). The following exposure paths are considered for the population groups, children (one year old) and adults: gamma-dose by submersion and deposition, inhalation, ingestion (milk, meat, leaf-vegetables, other vegetables).

5. Outlook

It is planned to realize the integrated MADUK-ANPA system by the end of 1993.


Nuclear Power Plant (NPP) Mühleberg
Nuclear Power Plant (NPP) Beznau
Nuclear Power Plant (NPP) Leibstadt
Nuclear Power Plant (NPP) Gösgen
Meteo-
Tower
Dose Rate
Monitoring System
(2 Detectors of 16)
Data Concetrator
Leased Lines
Meteodata
Wide Aerea Network
Swiss Telephone Company (PTT)
Swiss
Meteorological
Institute (SMA)
Zürich
Telepac
Data Concetrator
Würenlingen
Server
Zürich
Swiss Emergency
Center (NAZ)
Workstations
Swiss Federal Nuclear Safety Inspectorate (HSK)
Würenlingen
Section SUER
Federal Office of Health
Fribourg
Figure 1
The Swiss MADUK-ANPA Projekt for Dose Monitoring around the NPP's (MADUK) and for Emergency Response Data Transfer (ANPA)
Hardware Configuration
MA1707B5/HSK/UP



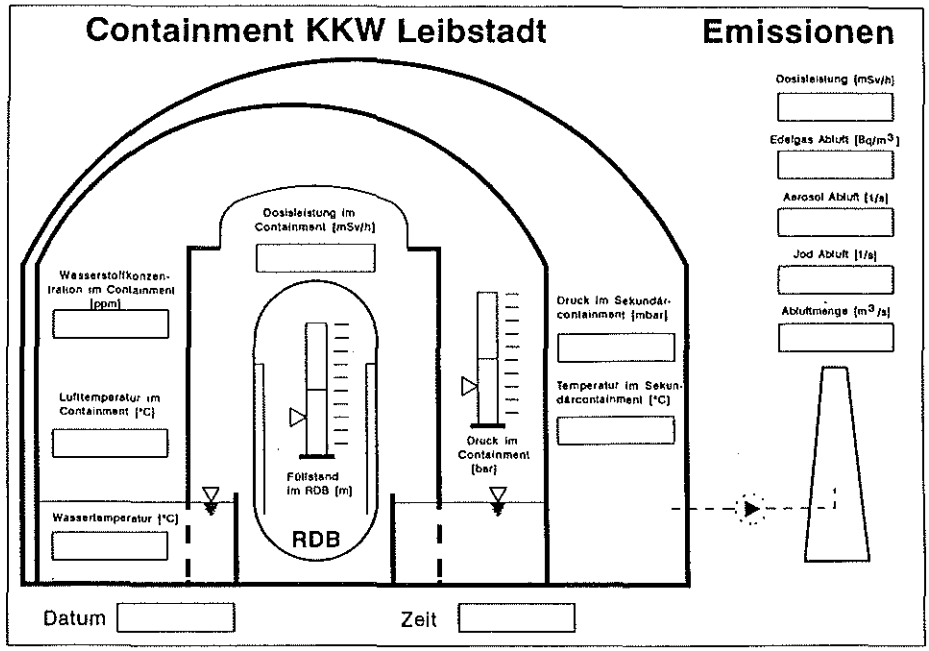


Figure 2 The Swiss MADUK-ANPA Projekt for Dose Monitoring around the NPP's (MADUK) and for Emergency Response Data Transfer (ANPA)

2 Examples of ANPA-Graphics

MA1707B4/HSK/UP

VISUALISATION OF RADIOPROTECTION DATA AT NUCLEAR POWER PLANTS BY MEANS OF A GRAPHICAL USER INTERFACE

Authors:

Author: Dipl. Betriebswirt (FH) **Carsten Fischer**
Co-Author Dipl. Ingenieur **Stephan Pawlytsch**

Company:

Admintec GmbH
Bayernstraße 53
W-7530 Pforzheim
Germany

Phone: (0) 7231 / 38 08 65
Fax: (0) 7231 / 38 08 57

Summary

The Radiation Protection Atlas is a database program to assist the radiation protection department of a nuclear plant.
All stored data can be graphically shown on a monitor, e.g. the values of dose rates of a measuring point in a room sketch can be shown as figures or illustrated in colours. Therefore, all data are available in no time and easily surveyed for any planning and precaution. The Radiation Protection Atlas helps to ensure the quality of the radiation protection.
Additionally, the Radiation Protection Atlas allows to collect, to administrate and to show the measured values of the environment.

1. Information and Planning System for the Radiation Protection at Nuclear Plants

The complete control area is stored in form of building drafts. Related to these building drafts all important information such as activity values, additional detailed information and pictures can be stored.
The goal of this system is to assist the radiation protection as the stored data can be quickly found. As opposed to a conventional data collection such as lists and incompatible application platforms, the data can be used very flexible for a concrete task.

Due to the modern software technology the program is easy to learn. The program supports the user with an intuitive and self explaining user interface.

Any parts of the building drafts can be shown on the monitor using the whole screen. This means that small pictures are zoomed until they fill the whole area.
The mouse is used to select elements such as a measuring point, a component or a wall. Then the detail information of these elements is shown in a window on the screen.

Detail Information: e. g.

- Course of dose rate of a measuring point
- Detailed drawing of a component
- Technical data of a component

Depending on the task the representation form of the data can be selected from the following list:

Representation Forms: e. g.

- Coloured code for the measure value at any time
- Standard symbols of the components
- Exact outline of components

Information shown on the monitor can be printed or can be used from other programs for further manipulation.

Thereby, unnecessary radiation exposition or contamination of persons, hardware and the environment will be avoided during the planning and realisation of radiation protection tasks.

2. Operational Areas

2.1 Planning of Work

- Activity planning to avoid uptaking in the body and expansion of contamination.
- Pre-examination of the basic marginal conditions such as local dose rates, contamination and local conditions.
- Estimation of the expected individual doses.
- Access to the existing experiences of the radiation expositions

2.2 Conducting the Work

- Drawing pictures and establish measuring records as an enclosure to the working plan

2.3 Documentation of the Work

- Proceedings of the dose rating.
- Collective dose linked to the orders.
- Documentation of the possibly used radiation protection activities as well as a documentation of the essential data of the radiation protection oversight.
- Estimation of doses, the planning of the dose rates, the planning of the room organisation as well as additional information.

2.4 Annual Planning, Annual Survey and Statistics

- Preparation of a forecast including the estimated annual collective dose rate for all the planned maintenance- and service work.

3. Functions

3.1 Evaluations

- Analysis and statistics for the most important applications.
- Export of information to word processing programs.
- Sharing of information with spreadsheet programs with the possibility of graphic evaluation.

3.2 Standard Operating Methods

For often repeated operations are "standard-methods" available.
Orders to measure dose rates (air activity, contamination, ...) can be planned with a computer. With this planning all necessary order forms for measuring will be automatically prepared.

3.3 Make an Alteration

Normally any modification in conventional plans needs to be changed in all plans related to this alteration. Examples are floor plans, room plans, and component lists. In contrast to this conventional way, only one alteration has to be done with the Radiation Protection Atlas.

3.4 Research Functions

All information stored in the Atlas can be represented at the monitor at any time. Special search functions such as catalogues for catchwords, indices and hierarchies allow the retrieval and presentation of the desired information within seconds.

3.5 Forms and Reports

- Operating sketches for the radiation protection can be printed including the existing plans and information which can be - if necessary - modified with a graphic editor.
- Radiation protection orders are created with text modules and the existing data from the Atlas.
- The user can use the data from the Atlas for the creation of reports and documentation for the bof control. This data can be printed in lists or reports using user defined criteria.

3.6 Data Import and Export

It is possible to import data like graphics and text files from other programs into the Radiation Protection Atlas.
For the preparation of individual reports the data can be exported to other programs like spreadsheet and word processing programs.

3.7 Authorisation

The Radiation Protection Atlas distinguishes various kinds of hierarchic user groups.

3.8 Detailed Information

The data from objects can be linked to additional detailed drawings, text documents, graphics and digitised photos. This enables the user to deposit permanent detailed information into an existing

object at any place. These data can be shown in windows at any time by clicking the corresponding icon with the mouse.

3.9 Planning of Doses

Individual and collective doses will be transferred from an existing database. These doses arise during certain activities and orders. These doses are used for the planning of comparable activities.

3.10 Documentation of Photos

Collection and administration of digitised photos and their connection with other data.

3.11 Radiation Protection Permits

All the radiation protection permits will be planned with the PC. At the time of the realisation of the order the data will be printed on standard forms with all enclosures (Sketches and detailed instructions).
Old radiation protection permits can be retrieved and - if necessary - modified.

3.12 Reduction of Doses of Work Exposed

With a simple arithmetic module weighted dose rates are determined of different overlapping dose rate areas in rooms for locations of interest.
In addition to this the module allows the graphical representation of the progress of iso-doses curves.

- Planning of the expected individual and collective doses
- Shielding calculation
- By using old measured values of a component the optimal dose rate value of a component is determined with the computer (For example filling position, degrees of contamination and cleaning).

3.13 Control and Information Systems

The Radiation Protection Atlas can also be used as a control and information system during the operation and during the revision/stop (For example multiple shifts at the radiation safety).
Time consuming paper documentation are not needed, the transport of the necessary information to higher organisational units or control places occurs without delay.
The input of data can be done manually or using a notebook up to the connection of notebook with dose rate measure systems. On certain conditions the Radiation Protection Atlas can replace the "shift book".

3.14 Measurement with Automatic Measurement Value and Measurement Point Data Collection

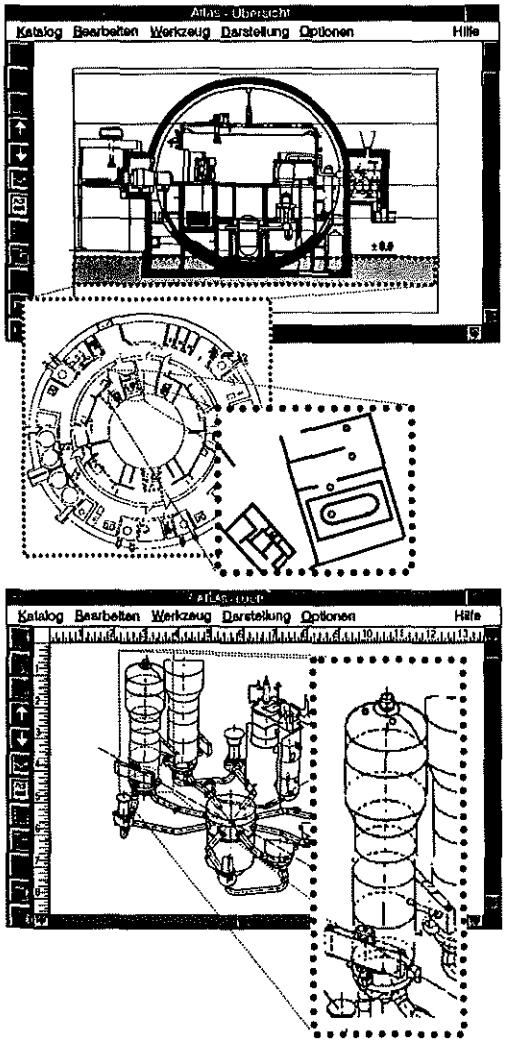
Measured values with an automatically registered meter and its measuring point, which will be registered with a barcode reader, will be processed with this module.

3.15 Supervision of the Environment

- Automatic recording of all environmental measured values
- Reports
- Continuous collection of measured values

4 Hard- and Software Requirements

- IBM Compatible Personal Computer (at least 80386)
- Microsoft Windows 3.1



REAL TIME MONITORING OF THE ENVIRONMENT AT THE NUCLEAR RESEARCH CENTER AT *FONTENAY-AUX-ROSES*

T. PINEIRA, J. GAUDIAU

Commissariat à l'Energie Atomique - Centre d'Etudes Nucléaires de Fontenay-aux-Roses - Service de Protection contre les Rayonnements et de l'Environnement - BP 92265 - Fontenay-aux-Roses CEDEX

Summary

The level of radioactivity at the Nuclear Research Center at Fontenay-aux-Roses is monitored on a continuous, real time basis. A star-configuration network, with two security levels, is used to collect file, manage and centralise the data generated by the sensors.
The Environment Control Panel (TCE) contains the computerised control resources and ensures two types of monitoring functions. The first monitoring function is preventive in nature and is intended to limit or stop any release not corresponding to authorised levels. The second monitoring function provides a continuous check of the radiological impact of the center's activity on its environment.

A computerised and centralised control of the malfunctions and alarms is used to facilitate the on-going round-the-clock monitoring function.

This system is linked with the SYTAR network connecting the CEA and EDF centers, thus contributing to the radiological monitoring of the french main land.

Introduction

The level of radioactivity in the environment at the Commissariat à l'Energie Atomique sites has been monitored since their creation. This monitoring function is organised and directed by the Management at each center which implements the resources required to observe the strict regulations relative to protection of the environment and population. These regulations are covered by french administration laws passed in MARCH 1988 specifying, for each center, the conditions for disposal of radioactive liquid and gaseous wastes [1] [2].
Radioactivity monitoring must take place on a real time, continuous basis, generally requiring installation of a computerised monitoring system for acquisition, control, centralisation and filing of data received from the sensors. This document describes the various components and functions of the environment monitoring system at the Centre d'Etudes Nucléaires de Fontenay-aux-Roses.

1. Purpose of environment monitoring system

The purpose of the environment monitoring system at the Centre d'Etudes Nucléaires de Fontenay-aux-Roses (CEN.FAR) is to centralise, in real time, the data generated by the sensors placed in :

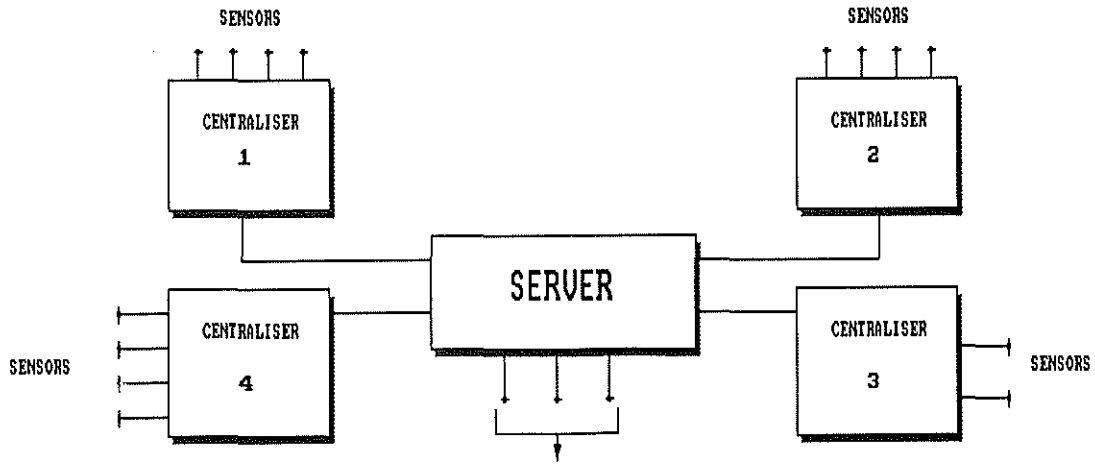
- chimneys of the basic nuclear installations (INB),
- the liquid release control stations,
- the radiological control stations outside the Center,
- the weather station.

This configuration ensures two types of control :
- control before and during releases,
- environmental control.

The first control function is preventive in nature and should be capable of limiting or stopping a release not corresponding to authorised levels. The second control function provides an on-going verification of the radiological impact of the Center's activities on its environment.

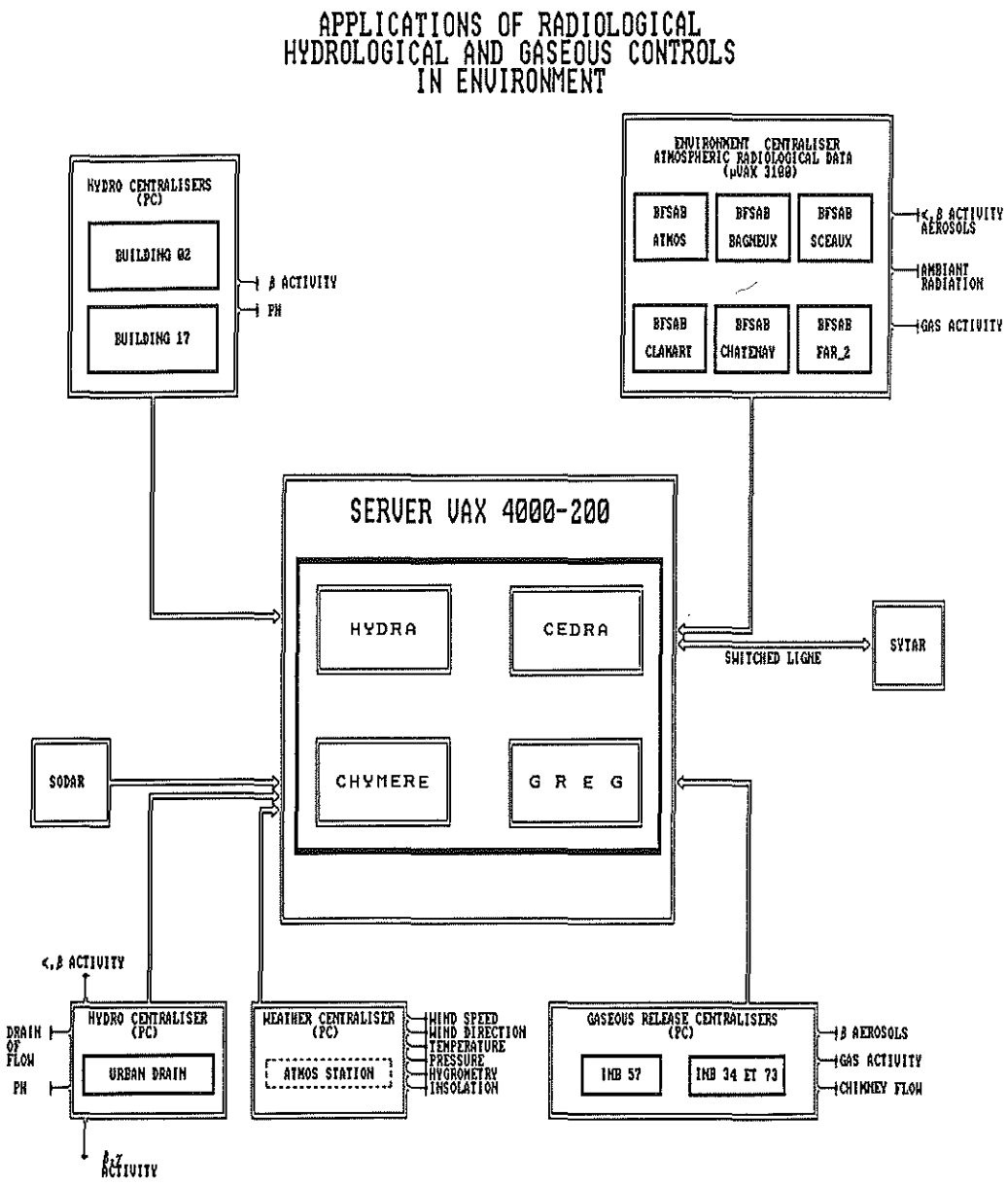
2. General architecture of computer system

The chosen schematic principle (figure 1) is a star-configuration network with two security levels. Figure 2 gives a general block diagram of the system.



THE SERVER IS A VAX 4000 MANUFACTURED BY DIGITAL
TWO TYPES OF CENTRALISERS ARE USED, μVAX 3100 OR PC COMPATIBLES

CEN-FAR
SPRE-SRSE
Figure 1



CHYMERE : MANAGEMENT OF HYDROLOGICAL AND METEOROLOGICAL DATA OF ENVIRONMENT SYSTEM.
CEDRA : MANAGEMENT OF RADIOLOGICAL, ATMOSPHERIC DATA IN ENVIRONMENT.
HYDRA : MANAGEMENT OF HYDROLOGICAL DATA IN ENVIRONMENT.
GREG : MANAGEMENT OF GASEOUS DISCHARGES.

CEN-FAR
SPRE-SRSE
Figure 2

3. Real time control of releases

3.1 Gazeous releases control

Specific sensors are used to continously measure the beta aerosols activity, gas activity and flow in the chimneys of INB 34, 73, and 57 at the Center.
These sensors are connected to two centralisers locally handling the information and transmitting it to the VAX server every ten minutes.

3.2 Liquid releases control

A liquid release is performed when the radiological and chemical analysis of the sample taken in the tank satisfies the authorised levels.
Two stations, HYDRO 02 an HYDRO 17, continuously monitor the beta activity and the pH of the radioactive effluents released by the two specific sectors of the Center.

A third HYDRO station, located downstream of all the release points, monitors all of the effluents released into the urban drain.

The HYDRO station is equiped with :

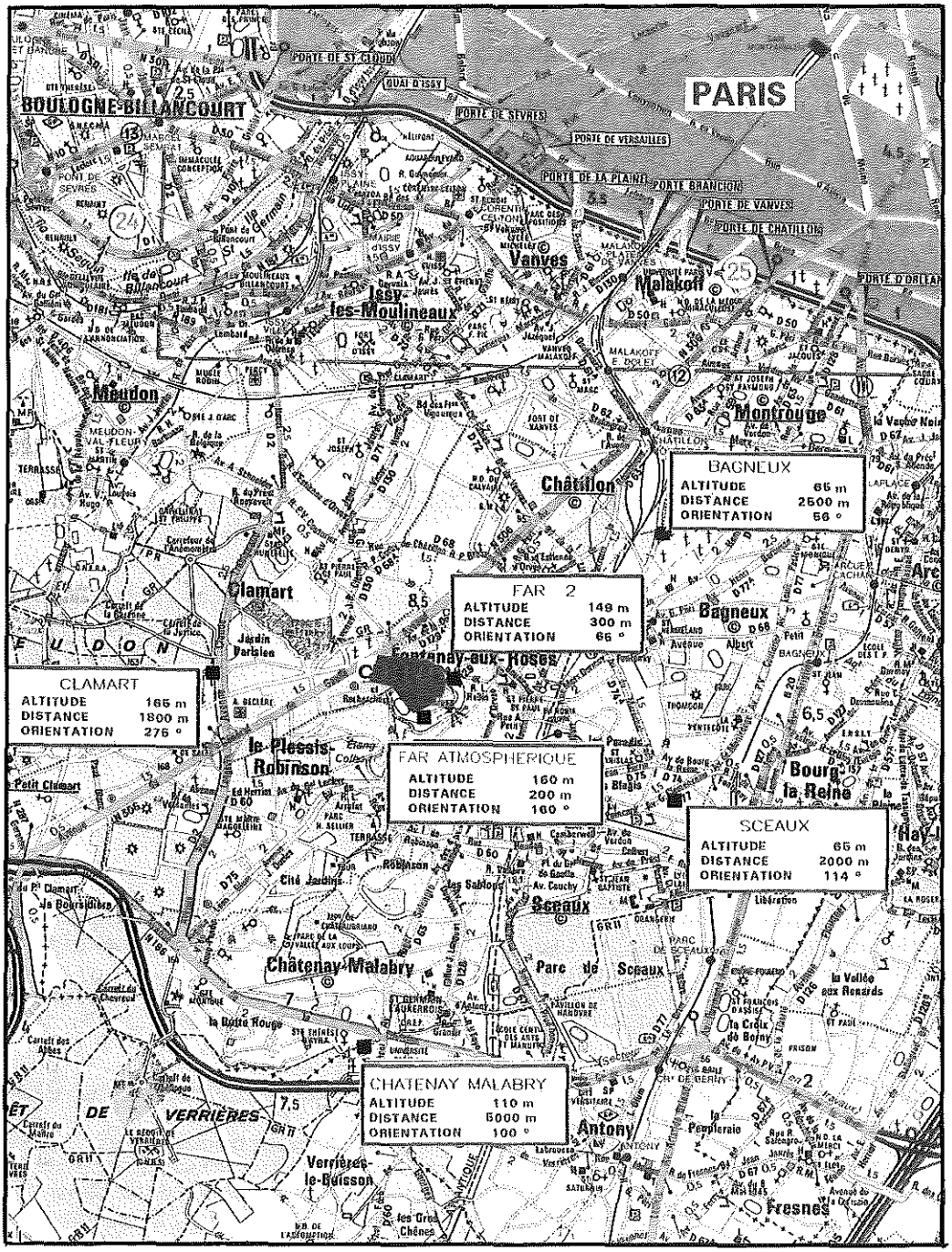
- a device continuously controlling the alpha and beta activity (Rame 11),
- a device continuously controlling the beta and gamma activity (Cobenade),
- a pH controle device,
- a flowrate measurement device in the drain.

These measures are centralised on two PC compatibles then retransmitted to the environmental control panel. CHYMERE software ensures acquisition and management of all the data.

4. Real time control of the environment

4.1 Radiological control stations

Monitoring of the atmospheric radioactivity is ensured by six radiological control stations with layout given in figure 3. Each station is equipped with a beacon (BFSAB) measuring the alpha an beta activity of the aerosols. The associated electronics is used to dissociate, in real time, the natural activity due to the radon-thoron daughters and any possible artificial activity. This information is completed by measurements of the ambiant radiation and of the activity of the gaseous component of the air given by a large volume differential chamber (CD 43).
BFSAB processor is connected to the VAX 4000 through a centraliser (μVAX 3100). CEDRA software [3] ensures acquisition and management of all the data collected.



MONITORING STATIONS OF
FONTENAY AUX ROSES RESEARCH CENTER

CEN - FAR
SPRE/SRSE
Figure 3

4.2 Measurement of meteorological parameters

The parameters indispensable to a prediction study of the releases, studies relative to impacts and atmospheric transfers are originated by two installations :

weather Center station : (coupled to the "ATMOS" atmospheric control station) equipped with a 30 meter mast use to measure wind speed and wind direction, temperature and temperature gradient, atmospheric pressure, hygrometry and insolation.
This information is centralised on a micro-computer which also ensures retransmission to the VAX 4000 server.
The data is acquired and processed by the CHYMERE software.

SODAR, which is an "acoustic radar" located within the Center and used to evaluate the meteorologic conditions between 50 and 400 meters ; the data is centralised on a mini-computer (PDP) and transmitted to the VAX 4000 server every 15 minutes.

5. Environmental control panel (TCE)

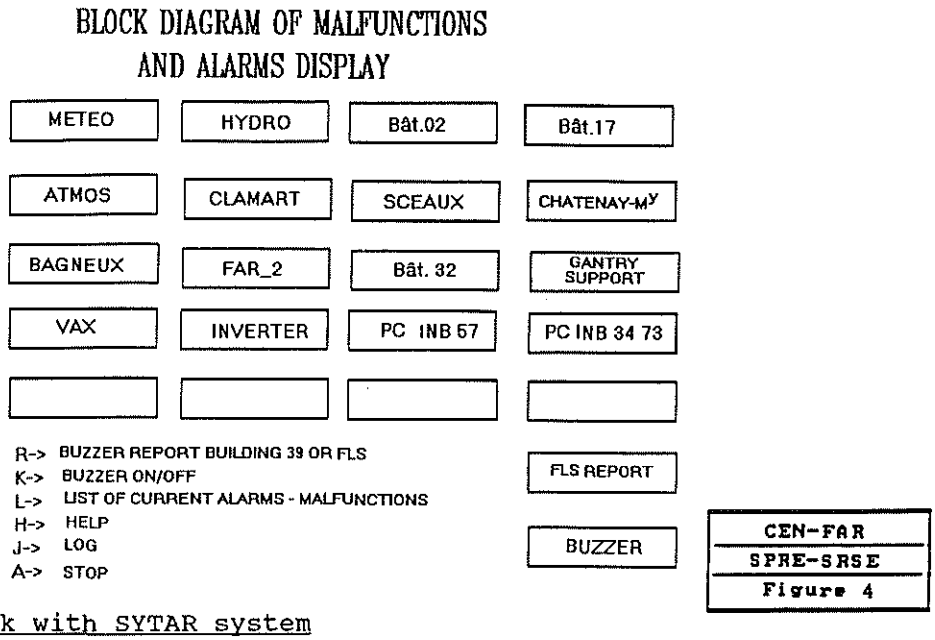
5.1 Description

The VAX 4000 server is a real time, multitask, multiuser system, used with the VMS operating system.
Each centraliser has a corresponding software performing data acquisition, storage and handling. Maximum security was defined for the filing function which uses two shadow disks.
The user interface, TCE software, is designed to facilitate access to the information. This interface provides fast transition from one control program to another, and from one specific data assembly to another.

5.2 Operation

Computerised control ressources : the display consoles, printers and some centralisers are grouped in a room called the Environmental Control Panel (TCE).
As a true environment control station, the TCE is also equipped with communication facilities : transceivers, telephones, fax, "Minitels".

Monitoring is ensured 24/24 hours through a round-the-clock duty system.
Malfunctions and alarms are handled on a micro-computer linked to the VAX 4000 server, to centralisers and to main operating components (inverters, ...). The centralised malfunctions and alarms data is displayed on a computer screen (fig. 4). In normal operation, all the indicators are green ; detection of a malfunction or alarm results in a change of colour, respectively to yellow or red. A buzzer warns the user of the change of state.



6. Link with SYTAR system

The SYTAR system [4] is a radiological transmission and alarm network. This system was created in response to the need for precise and fast information which became evident since the accident which occured at the TCHERNOBYL power plant.
This system covers the entire French main land and links the CEA and EDF Centers. Information transmitted is structured into four levels :

- level 0, nature and state of operation of equipment,

- level 1, measurement of atmospheric alpha et bêta radioactivity, external radiation and gas radioactivity : real time,

- level 2, alpha, beta and gamma spectrometry of aerosols, water and ground deposits : delayed,

- level 3, alpha, beta and gamma spectrometry of grass and vegetal samples (fruits, vegetables, leaves) : delayed.

The 0, 2 and 3 level data are manually entered and sent through a PC type computer linked by a switched line to the SYTAR server. The CEDRA software [3] provides real time transmission of the level 1 data. The VAX 4000 is linked to the SYTAR server through a switched line.
Associated to the air radioactivity files are the meteorological data which are transmitted in real time to the SYTAR server.
A SYTAR-MINITEL application provides the possibility to the public of access, in simple form, the results of the measurements performed around the Centres linked to the SYTAR system. These informations are avalaible on "MINITEL" 36 14, code "MAGNUC".

7. Conclusion

The environment around FONTENAY-AUX-ROSES is monitored in real time. The monitoring function is organised around a star-configuration computer network based on DIGITAL VAX 4000 and centralising data received from :

- the chimneys of the basic nuclear installations,
- the liquid release control stations,
- the 6 radiological control stations outside the Centre,
- the weather station.

The measurements relative to atmospheric alpha and beta radioactivity, external radiation and gas radioactivity are also sent, in real time, to the SYTAR network which combines and inputs the information received from the CEA and EDF Centers into a national radiological alarm system. SYTAR-"MINITEL" application provides the public with access to the results available on the server.

The Environment Control Panel (TCE) groups the computerised control resources, ensures 24/24 hours monitoring, and continously verifies the radiological impact of the Fontenay-aux-Roses Centre on its environment.

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THE MONITORING OF STRAY RADIATION FIELDS AT *CERN*

K. Goebel, M. Höfert, J.W.N. Tuyn and D. Wittekind
European Organization for Nuclear Reseach (CERN)
CH - 1211 GENEVE 23

SUMMARY **The extensive system of active and passive detectors for monitoring stray radiation levels on the CERN site and its boundaries is presented. It will be shown how the ICRU quantity "ambient dose equivalent" to be used for area measurements is introduced and the consequences of this implementation will be discussed.**

RÉSUMÉ **La mesure de rayonnement diffusé au CERN.**

Le système étendu de détecteurs actifs et passifs pour la mesure des niveaux de rayonnement ionisant sur le site du CERN et à ses limites est présenté. Il est montré comment la quantité physique "ambient dose equivalent" de l'ICRU est utilisée pour des mesures sur le site; les conséquences de l'application de cette quantité sont discutées.

1. Introduction

The extensive network of active and passive radiation monitors installed at CERN to assess the immission of stray radiation from the operation of the high-energy particle accelerators has been described previously [24, 10]. Whilst over the years the radiation detectors employed remained mainly the same, the original data acquisition and transmission system was replaced by a modern area controller device [22]. In the following the implication of a calibration of the monitors in terms of the new ICRU quantities will be discussed in view of the dosimetry of stray radiation fields as they are encountered at CERN.

2. The detectors

There are essentially two active detector types employed in the stray radiation monitor system at CERN: a high-pressure ionization chamber for the detection of photons and muons and a BF_3 counter surrounded by a moderator made of polyethylene, a so-called Andersson and Brown "rem" counter, for the measurement of stray neutrons. The high pressure ionization chamber of the type IG5/A20 is filled with 20 atm of argon and has a typical sensitivity of 14.4 nAh/R for a photon energy of 1 MeV. The sensitivity of the "rem" counters as used in the site stations is of the order of 1.3 counts per nSv for a spectrum of neutrons from a ^{238}Pu-Be source.

The passive measuring devices consist of ^{6}LiF and ^{7}LiF thermoluminescence detectors (TLDs) inside cylindrical polyethylene moderators of 12.5 cm diameter and 12.5 cm height, which integrate the above-mentioned radiation components over a period of one year [27]. Whilst there are 27 active measurement stations at the fences and an additional 11 distributed over the CERN sites, the layout of the passive devices with a total of 250 detectors is much denser. With the results of these detectors isodose curves are established annually reflecting CERN's radiation producing activities. The curves for the Meyrin site for the year 1991 are shown in figure 1 [23].

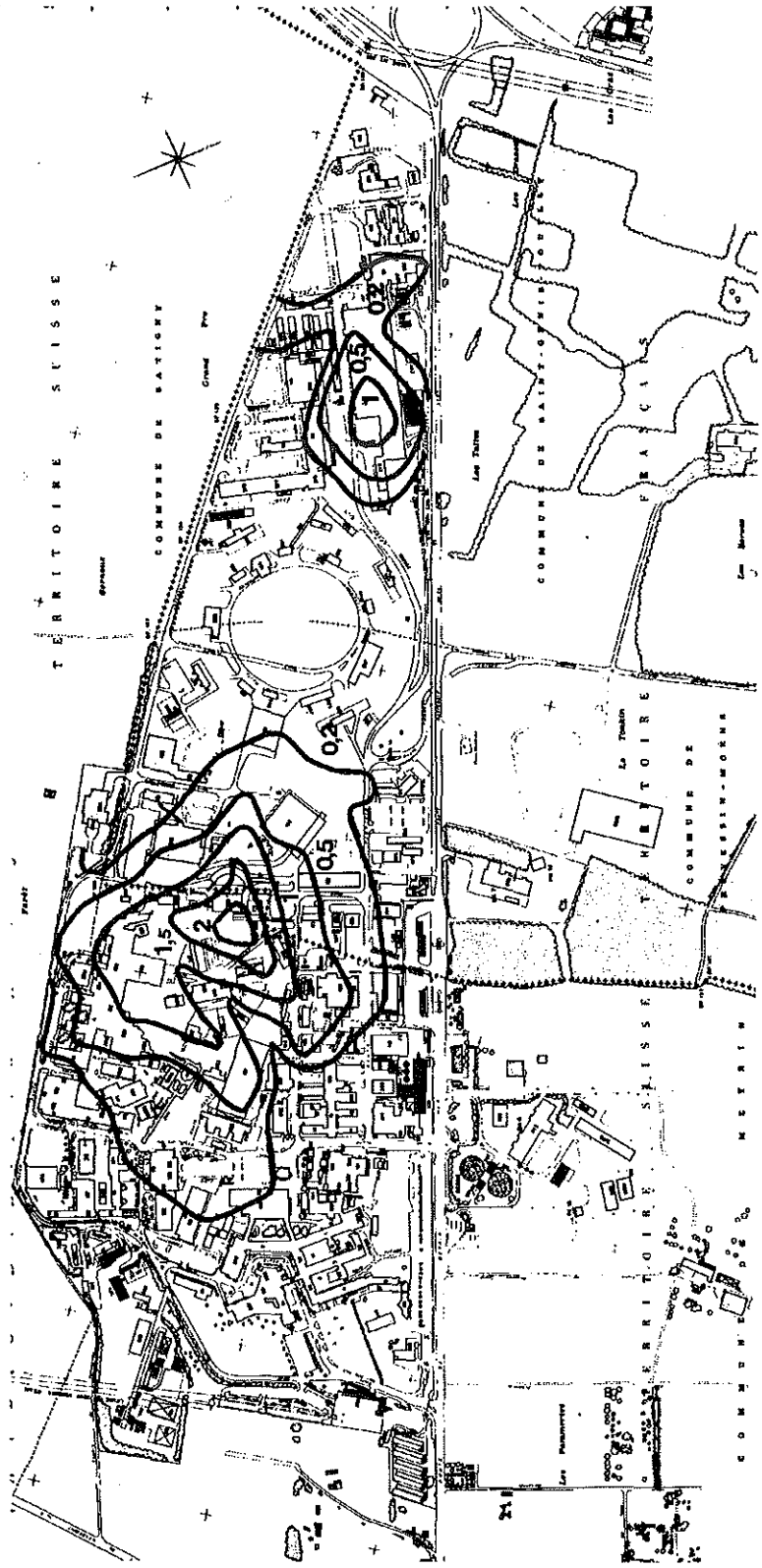

Fig. 1. - Isodose curves due to stray radiation on the Meyrin site of CERN for the year 1991. Dose values are given in mSv.

3. Quantities and units

For the specification of dose equivalent in environmental monitoring ICRU has proposed the quantity "ambient dose equivalent", $H^*(d)$, where d is equal to 10 mm for strongly penetrating radiation [16]. However, the actual intention of ICRU was not to limit the use of this quantity to the monitoring of only the environment, but to use it for area monitoring of stray radiation fields in general. This quantity will actually provide an estimate for effective dose equivalent, H_E, the risk quantity defined by the International Commission of Radiological Protection in Recommendation 26 [13]. In fact, it has been shown that $H^*(10)$ will overestimate both H_E and also particular organ doses H_T in all practical radiation situations, where in the case of neutrons these overestimations can reach more than one order of magnitude [17].

In the meantime ICRP has modified its definition of the risk quantity in Publication 60, i.e. in particular replacing H_E , the effective dose equivalent, by E , the effective dose [15]. At the same time, the radiation quality factors Q were replaced by the radiation weighting factors w_R. In addition to the fact that w_R are defined differently from Q the weighting factors are a factor of two higher than the quality factor for fast neutrons. All these changes are still much debated but when accepted will require the recalculation of nearly all factors that convert radiation field quantities into dosimetric quantities [2, 11]. The present paper will therefore concentrate on the concept of ambient dose equivalent, which is not going to change and will remain the universal operational quantity for area dosimetry irrespective of radiation type. In fact, it was shown that values of ambient dose equivalent calculated with the new weighting factors will actually give a better estimation for effective dose than values of ambient dose equivalent based on the old quality factors presented for effective dose equivalent [1].

4. Calibration

4.1 Photons

The active and passive detectors used for the monitoring of photons and charged particles in the environment on the CERN sites are usually calibrated with photons from a ^{137}Cs source. The quantity that was used up to now is called the photon dose equivalent, H_X, and is linked to exposure free in air, X, by the simple relation $H_X = c_1 X$, where $c_1 = 0.01$ Sv/R [25]. Photon dose equivalent replacing exposure free in air was proposed as an intermediate solution in radiation protection in Germany to permit the introduction of SI units until the more general quantity ambient dose equivalent, H^*, is introduced for radiation monitoring purposes and legally accepted. An advantage of photon dose equivalent is that one can continue using existing instruments for the measurement of exposure in radiation protection by simply scaling their reading by a factor of 100. In addition, the unit sievert for this quantity shows immediately that H_X is risk related. Other solutions to switch to SI units such as introducing the quantity absorbed dose with the unit gray in photon dosimetry are subject to confusion when it is not clearly specified whether air kerma or absorbed dose in tissue has been measured in a particular situation.

The energy response of the high-pressure argon chamber IG5 to photon dose equivalent normalized to ^{137}Cs gamma radiation as given by the manufacturer is presented in figure 2. For a calibration in ambient dose equivalent at the energy of 662 keV and for a depth of 10 mm in the ICRU sphere the following relation holds: $H^*(10) = c_2 H_X$, where $c_2 = 1.05$ [18]. This means that for a calibration of photon detectors in $H^*(10)$ all results of measurements will in the future be numerically 5% higher than those reported at present.

With the help of factors that convert photon dose equivalent into ambient dose equivalent as a function of energy the response of the high-pressure argon chamber to $H^*(10)$ has been calculated and is also given in figure 2. It turns out that the energy response of the chamber IG5 with respect to $H^*(10)$ is improved compared with its response to X in the energy range around 100 keV.

The relation between measured energy response and ambient dose equivalent for the cylinders (Cyl.) containing the TLD detectors is likewise given in figure 2 [28]. It turns out that these detectors have a

rather flat response to X-rays, varying within only ±20% between 30 keV to 1 MeV compared to the high-pressure argon chamber.

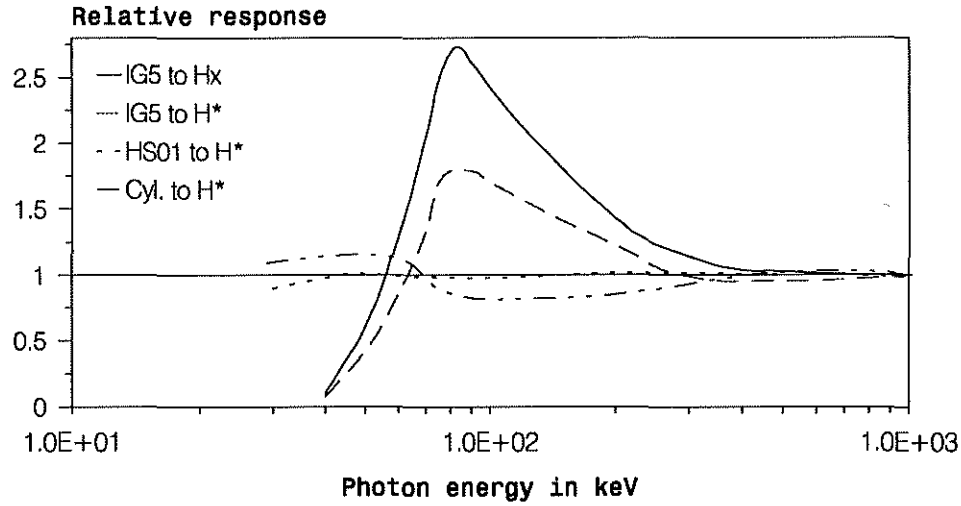

Fig. 2. - Relative response of photon detectors as a function of energy. Solid curve: Response of the high-pressure argon filled chamber IG5 to exposure free in air H_x. Dashed curve: Response of IG5 to ambient dose equivalent $H^*(10)$. Dotted curve: Response of the standard chamber HS01 to $H^*(10)$. Dash-dotted curve: Response of TLDs in polyethylene cylinders of 12.5 cm diameter and 12.5 cm height to $H^*(10)$.

Finally the response for a chamber (HS01) specially designed to measure $H^*(10)$ is plotted in figure 2 for comparison [7]. This chamber in its present form with a volume of only one litre is however not sensitive enough for use in environmental measurements; hence a more sensitive device is highly desirable.

4.2 Neutrons

Neutron calibrations of "rem" counters are traditionally performed with neutrons from radioactive sources. Whilst a calibration with neutrons from a ^{252}Cf source is appropriate for measurements in a reactor environment the use of moderator instruments around high-energy accelerators call for a calibration at higher energies for which neutrons from ^{241}Am-Be or at CERN from ^{238}Pu-Be sources are used. It is known that instruments for the measurement of dose equivalent based on the slowing down of fast neutrons in a moderator of polyethylene will overestimate in the intermediate energy range whilst the response of these instruments falls rapidly for neutron energies above 10 MeV [5]. Since this problem is of particular concern in the radiation environment around high-energy accelerators it will be discussed below.

In figure 3 several fluence-to-dose equivalent conversion factors for neutrons are plotted in the energy range from 100 keV to 20 MeV. The values marked ICRP 21 were recommended by ICRP in 1971 [12]. They are based on calculations of the maximum dose equivalent (MADE) in a semi-infinite tissue equivalent slab and are still widely used. Mean fluence-to-dose equivalent conversion factors for spectra from ^{241}Am-Be and ^{238}Pu-Be neutron sources were published as being 350 pSv cm^2 and 330 pSv cm^2 respectively [6]. Later the value for ^{241}Am-Be was revised by ISO to be 380 pSv cm^2 , but no value was given for ^{238}Pu-Be, the latter source no longer being considered an international standard [19]. The higher conversion factor recommended by ISO was subsequently adopted at CERN for both neutron source types.

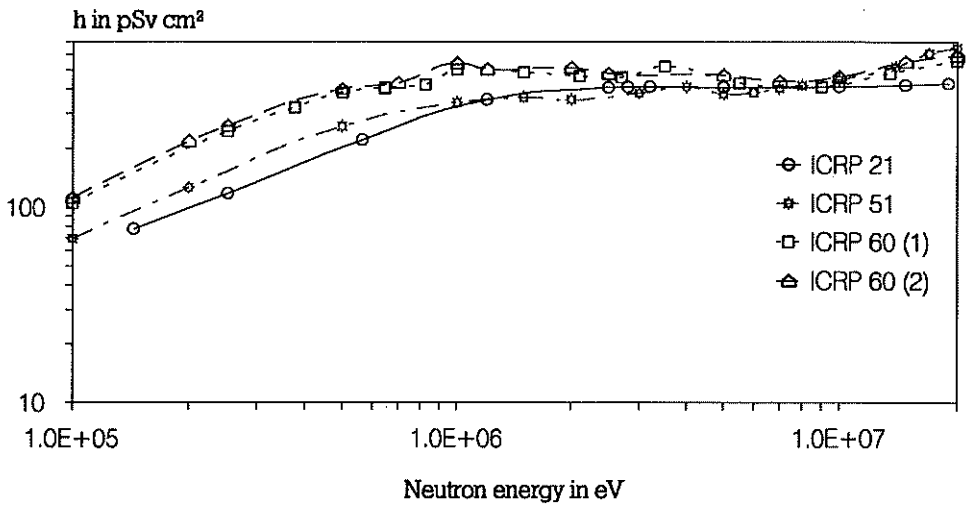

Fig. 3. - Fluence-to-dose equivalent conversion factors for neutrons as a function of energy. Solid curve: Maximum dose equivalent (MADE) as recommended in ICRP Publication 26 [13]. Dash-dotted curve: Ambient dose equivalent $H^*(10)$ recommended in ICRP 51 [14]. Dotted and dashed curves: Ambient dose equivalent $H^*(10)$ as calculated with the new radiation weighting factors for neutrons by Leuthold et al. [21] and Schuhmacher et al. [26] respectivly.

In ICRP Publication 51 fluence-to-dose equivalent conversion factors for the operational quantity ambient dose equivalent were presented [14]. Since in the neutron energy range above 1 MeV the differences between ICRP 21 and ICRP 51 are small, the conversion factor of 380 pSv cm^2 for ^{238}Pu-Be sources is still considered to be valid for $H^*(10)$ (figure 3).

With the advent of ICRP Publication 60 the new conversion factors for neutrons are now generally higher owing to the increase in the value of the radiation weighting factors. Recently two sets of fluence-to-dose equivalent conversion factors for the quantity ambient dose equivalent were calculated and published, and are plotted in figure 3 as a function of energy as ICRP 60 (1) and ICRP 60 (2) [21, 26]. Judging from the curves in figure 3 the increase in the conversion factors based on the new quality factors in Publication 60 for neutron sources having their fluence primarily in the energy range above 1 MeV should be less than 50%. In fact, a new conversion factor for ambient dose equivalent for neutrons from ^{241}Am-Be has been calculated and was found to be 465 pSv cm^2 compared with 380 pSv cm^2 previously [20].

As has been mentioned above, the response of instruments that measure dose equivalent based on the moderation of neutrons decreases rapidly at energies above 10 MeV. This behaviour could lead to an underestimation of dose equivalent when using such an instrument for the measurement of stray fields if higher energy neutrons contribute considerably to the ambient dose equivalent. Such underestimations are known and are generally found at CERN in measurements with moderator instruments made near to the source, i.e. in the vicinity of concrete shielding around accelerators and beam lines. It is then necessary to complete the acquisition of dose equivalent by using a set of multiple detectors covering the high-energy range [8].

The energy response of "rem" counters at higher energies was recently improved by introducing a cylinder made of lead into the polyethylene moderator that surrounds the thermal neutron detector [4]. In an experiment at CERN performed in a hard spectrum, i.e. a spectrum with neutrons well above an energy of 10 MeV, this instrument measured nearly double the dose equivalent found by a classical "rem" counter [3].

During this intercomparison a tissue equivalent proportional counter (TEPC) was likewise used. Event-size spectra of lineal energy were measured at different distances from the shielding wall that was the source of stray radiation [3]. They are shown in figures 4a to 4c. Whilst for the spectrum near to the wall events above 200 keV/μm due to heavy recoil particles as a result of high-energy spallation reactions are important (figure 4a) their contribution in a spectrum registered at 10 m distance from the source is clearly smaller (figure 4b). Finally, outside the experimental hall at a distance of 50 m these high lineal energy events are no longer detected: on a normal background spectrum extending to 30 keV/μm are superimposed event sizes around 100 keV/μm that are due to recoil protons from interactions of fast neutrons in the tissue equivalent detector (figure 4c). Therefore one can assume that the high-energy neutron component in the stray radiation field is degraded with distance from the source owing to multiple interactions. Hence measurements with a classical "rem" counter would give a good estimate of $H^*(10)$ far from the source for the neutron component. It is planned to support this assumption in performing intercomparative measurements in using the Andersson and Brown instrument and the one with the extended energy range on the CERN sites in parallel.

4.3 Muons

Muons are part of the stray radiation fields outside the shielding of high energy accelerators. They are essentially produced in the decay of pions and kaons. Their energy deposition is mainly due to electromagnetic processes similar to those of electrons. The response of various dosemeters to muons has been investigated in the past [9]. It could be shown that the dose in tissue equivalent material or eventually the ambient dose equivalent can be calculated from the absorbed energy in the detector by multiplying this dose with the ratio of mass stopping powers for tissue equivalent material to those for the detector material weighted over the muon spectrum. For typical spectra in vicinity of high-energy accelerators these correction coefficients are 1.03 for the high-pressure argon chamber and 1.10 for the TL detectors. In order to apply the corrections properly, the contribution of muons to the photon and charged particle doses registered by individual detectors must be known. However, since muons contribute generally only very little to the total ambient dose equivalent on the CERN sites compared with doses from other sources like natural background, the results of the detectors are taken as such.

5. Conclusions

Environmental detectors used at CERN will in the future be calibrated in ambient dose equivalent. The advantage of this decision is that both photon and neutron doses are expressed in the same quantity and can now be added together. In the past when results were reported in different quantities this was strictly not allowed. On the basis of ICRP Publication 26 the changes in reported doses will be 5% higher for photons when passing from photon dose equivalent (exposure) to ambient dose equivalent. There will be practically no change when switching from MADE to $H^*(10)$. With the higher neutron weighting factors as proposed in ICRP Publication 60 neutron doses will however be higher by 22%.

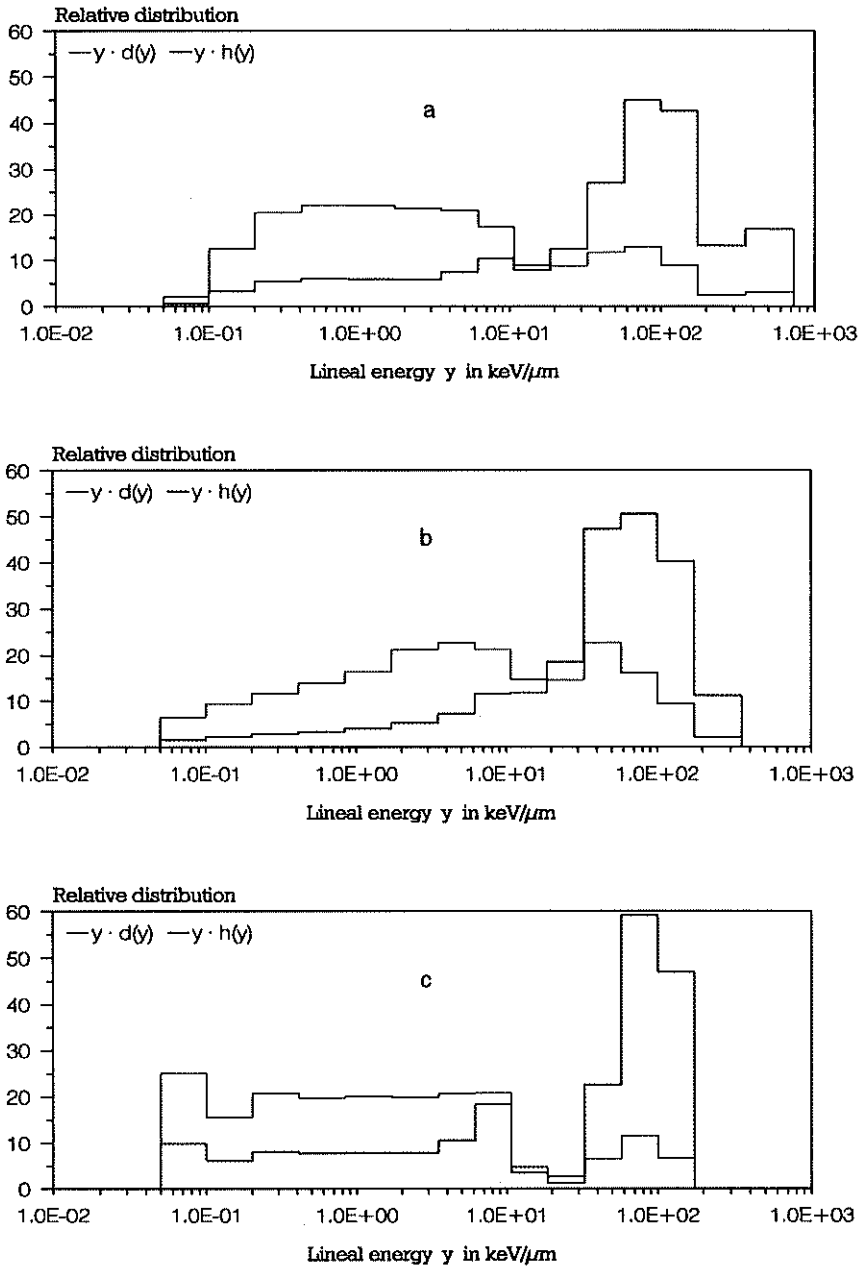

Fig. 4. Event size spectra measured with a tissue equivalent proportional counter. The distributions of absorbed dose $y \cdot d(y)$ and dose equivalent $y \cdot h(y)$ as a function of lineal energy y are presented for a: a measurement near the shielding wall, b: at 10 m distance from the shielding wall, and c: at 50 m distance from the shielding wall.

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RELEASE- AND ENVIRONMENTAL MONITORING OF THE *ATOMINSTITUT DER ÖSTERREICHISCHEN UNIVERSITÄTEN*

M.Tschurlovits, H.Slanetz

Atominstitut der Österreichischen Universitäten
Schüttelstraße 114 A-12 Vienna, Austria

Abstract
The methods used for release- and environmental monitoring of a university institute operating a reactor and other facilities are discussed. Results of the monitoring program are shown in brief.

1. General

The "Atominstitut der Österreichischen Universitäten" is an institute of the Technical University in Vienna and is providing facilities for research and training. In some of these facilities radiation is involved and monitoring is required.

The major facilities which may require release and/or environmental surveillance are:

- TRIGA Mark II research reactor
  - 250 kW steady state operation
  - 300 MW pulse operation
  - 4 beam holes
  - 2 thermal columns
  - facilities for radionuclide production
- 2 Van de Graaf generators
- radiochemical laboratories
- liquid waste treatment plant

Therefore, there is a potential that releases take place or a radiation level is enhanced. Some effort is therefore needed for release and environmental monitoring.

Although the potential of exposure is extremely low, legal requirements are fully applied. Therefore, the monitoring program in the present form was subject of approval by the competent authorities, although the existing program before licensing was only slightly changed in the licensing procedure.

2. Release monitoring

Fig 1. shows the site of the institute and the location of sampling sites

2.1 Airborne releases

2.1.1 TRIGA Reactor:

The major effluent is Ar-41 produced by neutron activation of air. Beam holes are exhausted into one of the two ventilation systems. Both systems are equipped with GM counters, and therefore release monitoring of the exhaust air is carried out. The data are recorded by a datalogger. The release rate depends obviously on the reactor power level, but also on the experiments in the beam holes. The activity concentration was determined as about 0,4 MBq.m^{-3} (1). Under present conditions, the release rate at full power is lower (2) as 0,1 MBq.m^{-3}, because in new experiments air volumes in beam holes are going to become filled with bulk material. This is leading to a normalized release of less than 5 GBq/MWh at present.

In addition, some air is bypassed in an air monitoring system for assessment of aerosol activity and iodine. The evaluation is carried out by Low Level beta-measurement (air filter) and gammaspectroscopy (iodine measurements). No radionuclides are detectable.

2.1.2 Chemical labs

The exhaust air of the radiochemical labs is monitored before release in a bypass system for aerosols. The activity on the filters is determined by liquid scintillation counting. This technique is chosen for high efficiency of counting alpha-emitters.

2.2 Liquid releases

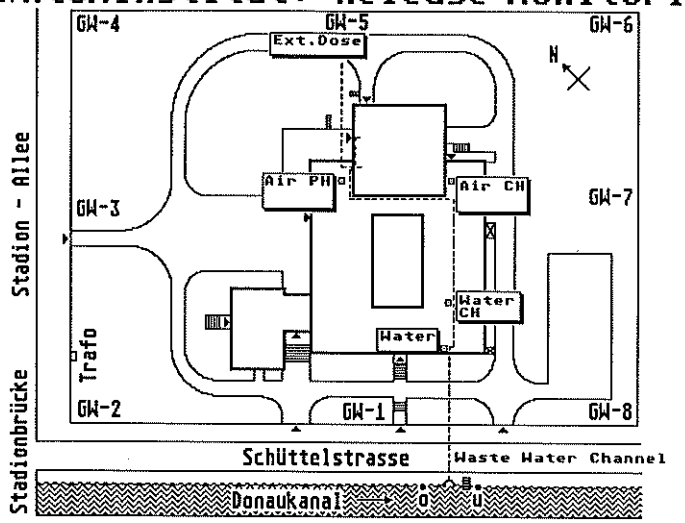
Possibly contaminated water from radiochemical labs is collected in storage tanks, allowing for decay. Water treatment is possible, if required. Immediately before release, a water sample is taken for measurement. After sample preparation by evaporation, gross beta measurement is done. If a given figure is exceeded, gammaspectroscopy is carried out. Release can be done only by dilution with reactor cooling water.
The total releases are in the order of 5 MBq/a and hence well below authorized limits.

3. Environmental monitoring

3.1 Sampling

Samples from various materials are taken in regular intervals. The sampling sites and the frequency are shown in fig 2.

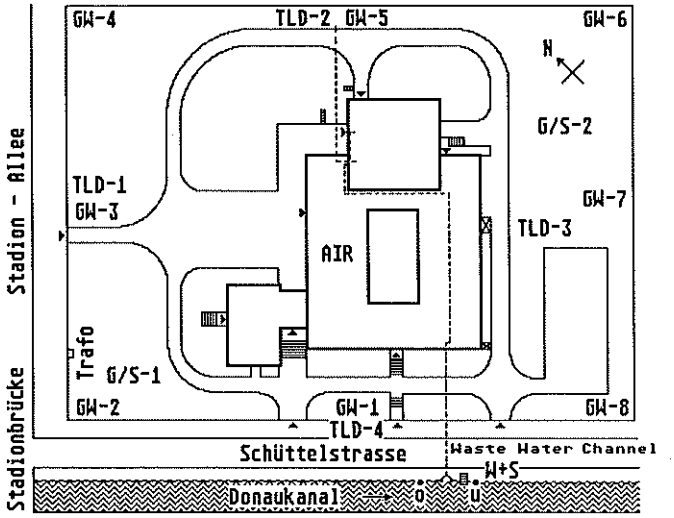
Fig.1: **Atominstitut: Release Monitoring**



Legend:

Ext. Dose	= External Dose Rate: continuously
Air PH & CH	= Exhaust Air: filters weekly; noble gases: continuously
Water CH	= Waste Water: at releases
Water	= Waste Water: continuously
GW-1 ... GW-8	= Ground Water: monthly or half-yearly

Fig.2: **Atominstitut: Environmental Monitoring**



Legend:

AIR	= Air Filters: daily
TLD-1 ... 4	= Thermoluminescence Dosemeters: half-yearly
G/S-1 ... 2	= Grass & Soil: half-yearly
GW-1 ... 8	= Ground Water: monthly or half-yearly
W+S	= Water & Sediments: weekly

3.2 Measuring Technique

Air is collected daily in regular intervals, activity measurements by low level beta measurements or, if required, by gammaspectroscopy

Water samples are taken from ground water (8 sites) and from the river Donaukanal, which is adjacent to the institute. After sample evaporation, the same measurement techniques as for air filters are applied

Vegetation: Grass, fruits and soil samples are taken and evaluated by gammaspectroscopy

External dose
A continuously operated measuring station detecting photons is located at the border of the institute site at the extension of a reactor beam hole. The full power reactor operation enhances the external dose rate for less than 10 percent, leading to an annual dose of less than 30 µSv. The main use of this station was the recording of the external dose rate after the Chernobyl accident. Neutron dose is checked twice a year with a rem-counter.

In addition, at 4 sites, TLDs (LiF) are exposed for six month. Since routine evaluation gave non satisfying results and too large uncertainty, an improved evaluation procedure is applied.

4. Conclusions

The monitoring program demonstrates the compliance the operation of the possibly radiation emitting equipment of the institute with present regulation, and no environmental consequences result, as expected. Since the institute is located in an urban area, even the fact that "no consequences" appear has to be proved for some reasons.
Fortunately, impressive results from environmental monitoring can not shown except the assessment of background, but the equipment was sucessfully used in detecting the plume from Chernobyl (3).
Regarding the effort in both manpower and equipment, it is intended to use sensitive simple techniques (as gross beta measurements) for routine monitoring to show that the activity concentration is well below limits. On the other hand, enhanced and more time consuming techniques can be applied, but only if required.

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AIRBORNE RADIOMETRIC SURVEY OF THE ENVIRONS OF THE SWISS NUCLEAR INSTALLATIONS

G. Schwarz, L. Rybach
Institut für Geophysik, ETH-Hönggerberg, 8093 Zürich

Summary

Helicopter surveys permit rapid evaluation with complete areal coverage of the terrestrial gamma radiation from natural and artificial radioisotopes in the topmost layer of the ground. The gamma spectrometric measurements (256 channels) are performed by a 16.8 liter NaI detector.

In order to identify possible radiation level changes around the five Swiss nuclear installations (four power plants and one research facility) the surrounding regions of each site are surveyed annually. In addition, regions with elevated natural radioactivity are mapped within the framework of the Swiss National Geophysical Survey. In case of accidents with radioactive material, or from debris of nuclear-powered satellites, the system would be used to locate the radioactive sources.

1 Introduction

Airborne surveys permit rapid evaluation of radiation levels of large areas. In inaccessible regions (topography, development), as for example the Swiss Alps, surveys with complete areal coverage are only possible from the air. Because of the larger ground clearance and the higher speed, the coverage per unit time of an airborne system is about 2500 times larger than of a comparable ground system. This is of key importance in radiological accidents. Although the costs for the measuring instruments and the flights are relatively high, the resulting cost per surveyed area is clearly lower than for a comparable terrestrial survey.

The method is without doubt the most efficient tool to locate lost radioactive sources (for example debris of reactor driven satellites). Another application is the monitoring of nuclear power plants to determine the undisturbed background for reference in case of accidents. Dose rate maps allow an estimation of the variability and the regional distribution of the terrestrial radiation, which is important in radiobiology, particularly in the discussion on the effect of low radiation doses.

2 Data acquisition and processing

The detector used for the survey consists of four 4"x4"x16" prismatic, thallium-doped, sodium iodide crystals (total volume 16.8 liters). Each crystal is equipped with its own photomultiplier tube (PMT). The spectrometer, with automatic gain control based on the K^{40} peak, covers the gamma ray energy range of 40 keV to 3000 keV with 256 channels. The spectrometer data are collected every second together with radar altitude, time, barometric pressure, outside air temperature and aircraft attitude by a personal computer. Positioning is done with the satellite navigation system GPS with a precision of 25 m. If no satellite signal is available a vertically mounted camera is used for flight path recovery. The data are stored on memory cards. During operation the system is controlled by a remote control console.

Since NaI-detectors have a relatively poor energy resolution the data processing is based on so called energy windows. The energy windows are centered on spectral regions of special interest. The energy windows have to be sufficiently separated from each other to keep interactions small. The artificial isotopes Cs^{137} and the Co^{60} are determined using energy windows centered at 660 keV and 1250 keV. An additional window, the total count window covers the complete spectrum and is representative for the total amount of gamma radiation. Especially sensitive to artificial radiation is the ratio of the low energy part of the spectrum (below 1400 keV) to its high energy part, the so called Man Made Gross Count (MMGC) ratio (Hoover, 1988).

The data processing first requires numerous corrections to the raw data (e.g. instrumental effects like drift and dead time are removed, aircraft and cosmic background subtracted, spectral stripping, normalizing to reference altitude). The corrected data are then interpolated, after adaptive filtering, to a grid for map production (pixel size 125 x 125 m). Maps of total count, individual windows, and the MMGC-ratio are routinely produced. 3D-representations can also be obtained. The complete processing software including map generation has been implemented on a personal computer (Schwarz, 1990) so that the data can be processed directly after landing of the helicopter.

3 From count rates to radioisotope concentrations

For the conversion of the measured detector-specific count rates to the corresponding radionuclide activity the detector sensitivity is needed. It was determined on the basis of in-situ gamma spectrometric ground measurements (Leupin, 1990 and Murith et al., 1990) carried out at several locations around the Beznau nuclear power plant and the Paul Scherrer institute as well as in the Magadino plain (TI).

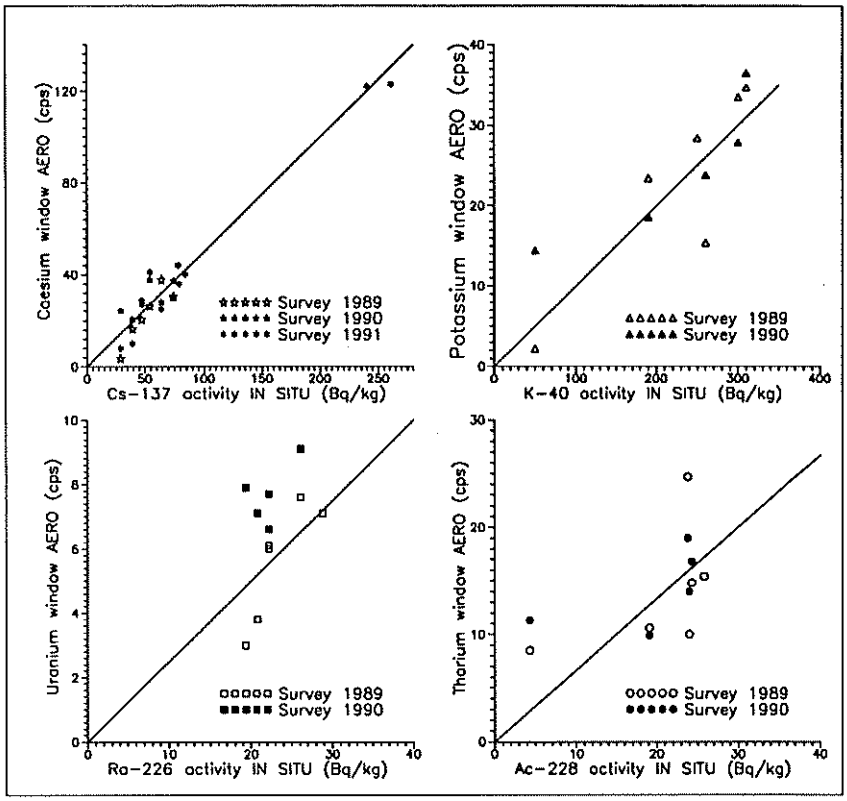

Figure 1: Experimentally determined detector sensitivities

The count rates measured with the airborne system are plotted against the insitu determinations of ground activity in figure 1. The ratio count rate/ground activity gives the detector sensitivity for the specific flight altitude (100 m in this case) for each energy window. Their values are: 50 cps in the caesium window per 100 Bq/kg Cs^{137}, 10 cps in the potassium window per 100 Bq/kg K^{40}, 31 cps in the uranium window per 100 Bq/kg Bi^{214} and 63 cps in the thorium window per 100 Bq/kg Tl^{208}.

4 Results

The regions surrounding (approx. 50 km^2) the four nuclear power plants (Beznau (KKB), Gösgen (KKG), Leibstadt (KKL) and Mühleberg (KKM)) and the Swiss nuclear research facility Paul Scherrer Institute (PSI) are surveyed annually since 1989. The airborne measurements are financed by the Swiss Federal Nuclear Safety Inspectorate and aim to monitor the dose-rate distribution and to provide a documented reference base (Schwarz et al., 1989, 1990 and 1991).

The flights where carried out with an Ecureuil helicopter of Heliswiss (Belp/BE). The flight altitude was 90 m, the flight line spacing 250 m and the flight velocity of 25 m/s.

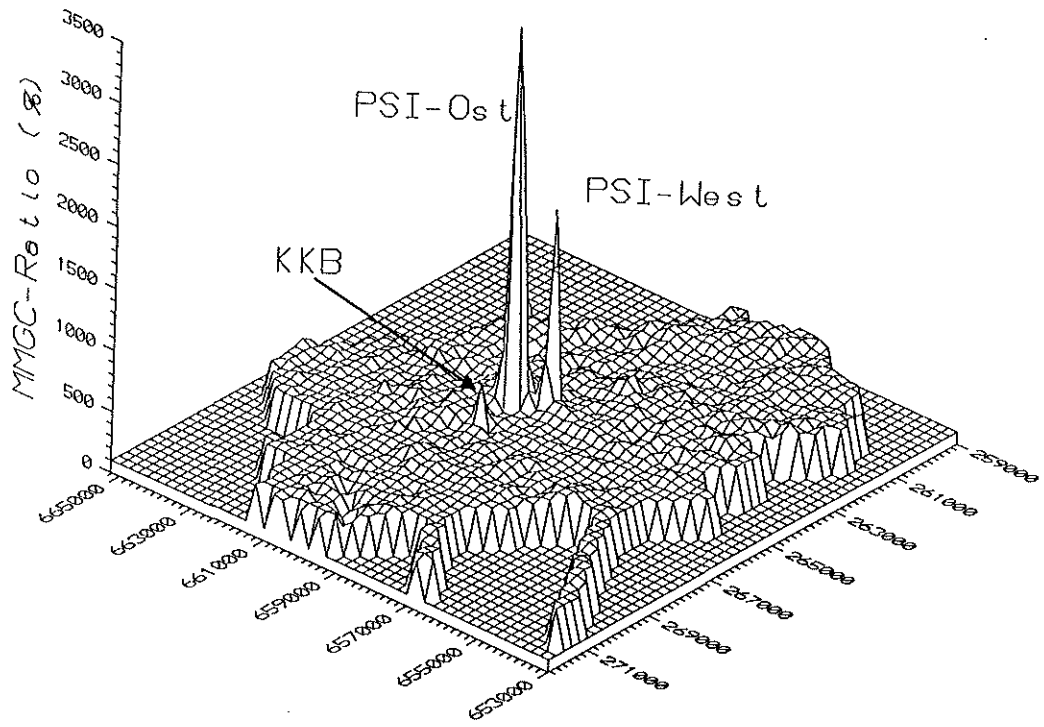

Figure 2: 3D-representation of the MMGC-ratio of the region of Würenlingen/AG (looking from north east).

Figure 2 shows a three dimensional representation of the MMGC-ratio of the surroundings of Würenlingen/AG (looking from north east). The two locations of the Paul Scherrer Institute PSI-East and PSI-West show up particularly well. The signal is caused by the direct radiation of the storage areas for radioactive components (PSI-West) and radioactive waste (PSI-East). The smaller peak is caused by the Beznau nuclear power plant.

All the other installations (with the exception of the KKG) also can be identified clearly on the MMGC-ratio maps. At sites of operating boiling water reactors the high energy radiation of the activation product N^{16} is clearly visible in the data.

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AIRBORNE MEASUREMENT OF RADIOACTIVITY

W. Dyck, H. Brust and E. Müller
Deutscher Wetterdienst, Kaiserleistr. 42, D-6050 Offenbach a.M.

Summery

The following is the description of an aircraft used for localizing and nuclide spezific measuring of radioactive contaminated air masses. For this purpose a high-volume collector without a pump for the collection of aerosols by filters, an on board gamma spectrometry system with a high purity germanium detector, a dose ratemeter, a radon daughter monitor and for navigation a global positioning system (GPS) are used. The meteorological parameters temperature, relative humidity and relative wind speed are also determined. The service ceiling is 33,000 feet. The resulting measurement data are used for verifying the prognosis made by the Deutscher Wetterdienst.

1. Introduction

After the Chernobyl accident, the Deutscher Wetterdienst requested measuring radioactive clouds according to their location, extent and isotopic composition up to an altitude of 33,000 feet (tropopause) in real time. The purpose of measuring the cloud is to provide and restart distribution modells with improved source data or to set them up again with the measurement data as initial and boundary values. The area of measurement, i.e. the area where the cloud is expected at the time of measurement, is taken from the (provisional) prognosis. This prognosis is either based on the plant managers information, measurements of other countries or at worst on ground measurements in the Federal Republic of Germany. The distribution models require information of the position and extent of the cloud as well as of the composition of the cloud's vertical section, if at all possible, vertically over the main direction of motion. The recording of radioactive clouds up to the tropopause is required, whereby the chances that transportation to altitudes up to 13,000 feet are most likely. The following measurements have to be carried out coincidently at various points in the atmosphere:

- gamma dose rate
- nuclide specific composition of the aerosol
- gaseous iodine
- temperature
- relative humidity
- relative wind speed
- coordinates and altitude

Measurements must be possible in real time in order to be permanently informed about the present location to the cloud and in order to adapt a flight route appropriatly in accordance with the meteorological experts. The aquired data are to be transmitted immediately to the Central Office of the Deutscher Wetterdienst in Offenbach, in order to achieve an improved prognosis as quickly as possible.

2. Method

Airborne measurement of artificial nuclides in the atmosphere was already being practised to a grat extent in the 1960s, whereby aerosol was collected on filters, which were measured for total beta activity and with gamma spectrometers only after landing /1/. After ratification of the atomic-weapon-block-contracts in 1963 which prohibited overground atomic weapon tests, these activities degreased. A new field of operation followed after the Chernobyl accident. Measurements with a germanium detector for measuring soil contamination were first realized by helicopter in autumn 1986 by the Bundesamt für Strahlenschutz in cooperation with the Reichsinstitut für Volksgesundheit of the Netherlands /2/.

The collection of aerosols by filters and direct measurement by a germanium detector during flight was first tested by the Deutscher Wetterdienst in September 1989 /3/. A germanium detector (relative efficiency of 20 %) was used in connection with standard electronic modules (NIM-electronics) and a portable multichannel analyser. Total weight, incl. lead shield (8 cm thickness of wall), rack and transformer (24 V/220 V conversion) was approx. 250 kg. Collection of aerosol was realised by a standard sampler with an air rate of approx. 3 m^3/h. The aircraft used was a Dornier, type 228 turbo-prop with non-pressurized cabin, owend by a company called Topogramm. Airport of operation was Dornier's works aerodrome in Weßling-Oberpfaffenhofen near Munich.

3. Development of the measurement system

After it was basically proved that operation of a germanium detector in an aircraft is possible, the Deutscher Wetterdienst developed a 19" compact measuring rack which can accommodate all the measuring equipment and can be set up quickly in various aircraft types (figures 1 and 2). It has to be as light as possible and provide all measuring systems with internal accumulators or 24 V board voltage. The air rate of the sampling system should be increased up to 500 m^3/h to reduce collection and measurig times and to im-

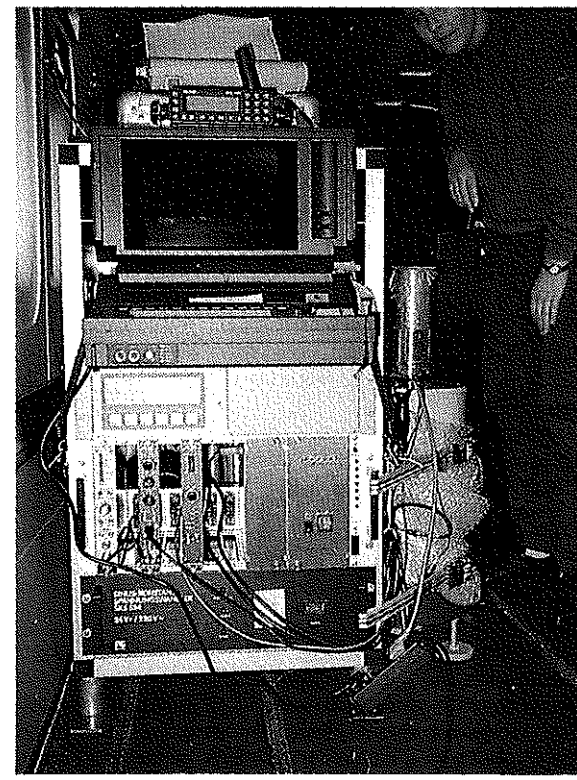

Fig. 1: Measuring rack (front)

prove the spatial resolution. In order to avoid operation of pumps with high electricity consumption, a special sampling system has been developed in cooperation with the Instrumentenamt of the Deutscher Wetterdienst in Munich-Oberschleißheim. Collection of aerosol works by airflow. The sampling system can be turned out of the cabin below the aircraft during the flight and sampling takes place practically isokinetically. In order to determine its rate, the air speed is ascertained in the influx tube via the Pietot and static pressure. The sampling system is turned back into the aircraft to change filter (fig. 3). A polypropylene filter of the 3M Company is used, which has a slight pressure drop by high collection efficiency (90 % according to DIN size 24 184) /4/. After collection, the filter is pressed into a reproducible geometry by a hydraulic press and then measured with the gamma spectrometer.

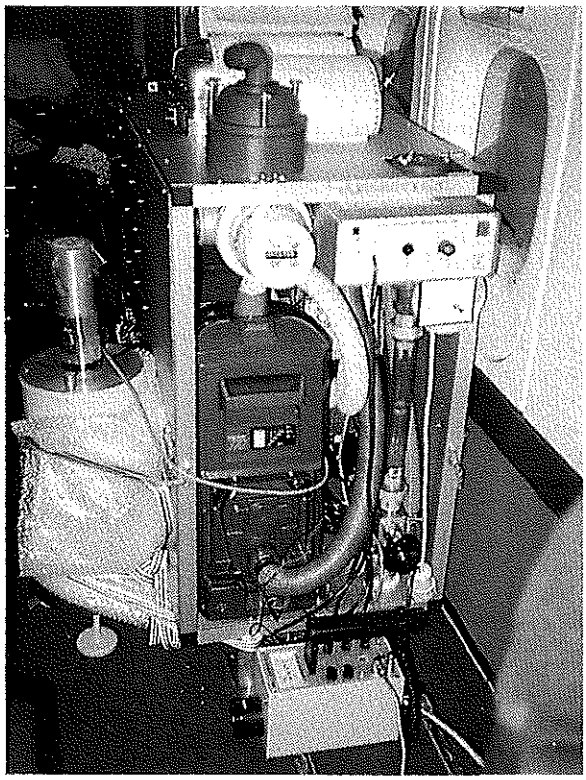

Fig. 2: Measuring rack (back)

The measuring rack contains following measuring systems, in detail (fig. 1 starting at bottom):
transformer 24 V/220 V for the operation of an additional pump when the measuring rack is used in smaller aircraft types where the high volume sampler can not be turned out; NIM-Bin with either 24 V or 220 V operating voltage and standard electronic NIM modules; battery operated gamma rate meter with data memory; on board laptop computer with multichannel analyser card for the registration and evaluation of the gamma spectra, 24 V operating voltage; battery-operated Global Positioning System (incl. navigation computer) to determine the coordinates, the altitude and relative wind speed.; battery-operated radon daughter monitor /5/; battery-operated ink printer; next to the measuring rack the germanium detector (40 % rel. efficiency).

(Fig. 2): sampling system with pump for max. 40 m^3/h air rate (standard hoover 220 V); on top of that a quantometer for determining the air rate; the pump is moved up slowly via thyristor control (above right); part of the exhaust air can be conducted via an iodine cartridge by means of a bypass (above right in the measuring rack); the air rate is determined by a

flow meter. Not visible in the photos: the proportional counter tube of the gamma rate meter; a sampling tube which can be extended telescopically through an outlet in the floor; airborne sonde for measuring temperature and rel. humidity, developed by the Instrumentenamt of the Deutscher Wetterdienst. This sonde is extended together with the sampling tube. An additional gamma rate meter is positioned in front of the rack. The total weight of the measuring rack without lead protector is approx. 100 kg. A lead protector of 50 kg (5 cm thickness) is available if required.

All data are registered by a laptop computer, converted into a homogeneous transmission format and transmitted directly to the Central Office of the Deutscher Wetterdienst in Offenbach via portable telephone.

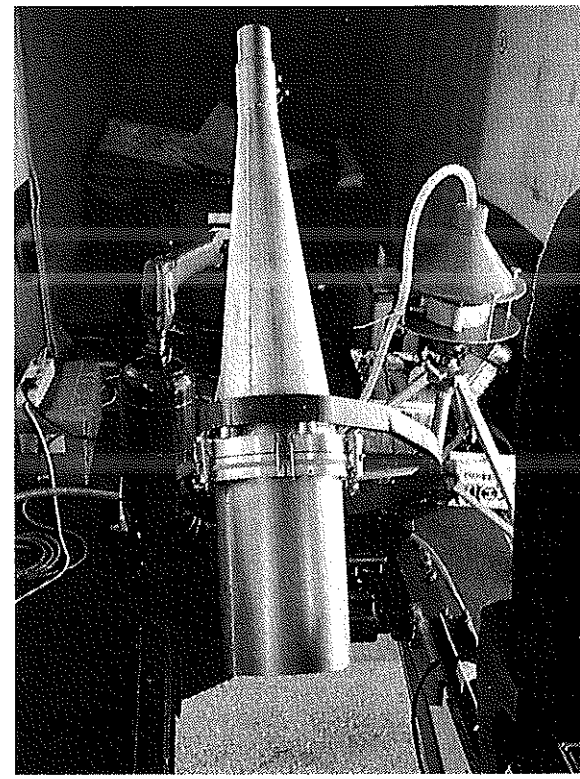

Fig. 3: Sampling system

4. Measurements and results

The operational qualification of all measuring and sampling systems has been tested during several flights, mainly in the Bavarian area.

It takes one hour to set up the rack incl. sampling system. All systems worked satisfactorily during the test flights. Aerosol was collected by filters at intervals of 3.000 feet (1000 meters) up to an altitude of 19.800 feet.

Artificial nuclides were not detected. The calculated detection limit for Cs-137 with an air rate of 420 m^3/h and a measuring time of 1.5 hours (40 % rel. efficiency) is 2.2 mBq/m^3. The registered gamma dose rate corresponded to the well-known dependence of altitude.

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DETERMINATION OF THE CONTENTS OF ALPHA EMITTERS IN VOLUMIC SAMPLES BY LONGTIME, LOW-LEVEL GAMMA SPECTROSCOPY

D.E. Becker, O. Mugrauer, K.D. Wünsch
Zentralabteilung Strahlenschutz
Technischer Überwachungsverein Bayern-Sachsen e.V.
8000 München 21, Westendstr. 199, Tel 089/5791/1558

Summary

To release material from nuclear facilities which might contain Thorium, Uranium or Transuranium nuclides its contents must be lower than 10^{-4} of the exemption levels of the radiological protection ordinance per gram of material, typically 0.5 Bq/g, and this must be approved. A new Gamma-detector device is presented for this purpose with a background countrate of 200 counts per keV and 500000 seconds in the range of 30-60 keV. The lower limit of detection for Americium 241 and for Uranium 235 is about 2 mBq. It is shown that the device meets the requirement.

1. Problem

To reduce the amount of nuclear waste, material which is not expected to be contaminated can be disposed of if the amount of radionuclides is lower than 10^{-4} of the exemption levels of the radiological protection ordinance /1/ per gram of material and if this is approved. Since the exemption levels for most alpha-emitters are 5000 Bq this results in 0.5 Bq per gram. This concentration is usually measured by radiochemical preparation followed by alphaspectroscopy - a rather expensive and time-consuming method. An alternative is the measuring of low energy gamma rays (20 - 200 keV). Since the gamma emission rate of most nuclides in this paper are in the order of some 10^{-4} a high-efficiency detector is required, combined with very low background.

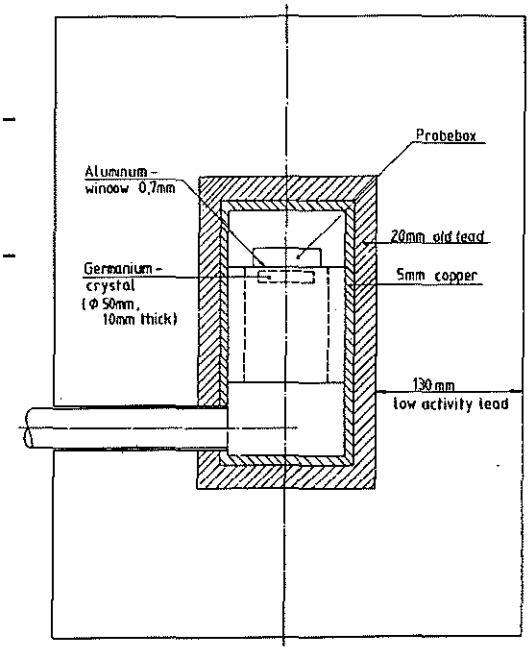

Figure 1: Detector inside the lead castle

2. The Gamma-Spectrometer

Background is reduced here by installing the detector in the second (lowest) basement so that six cover layers equivalent to 2 meters of concrete protect against cosmic radiation.

The surrounding lead shielding of 15 cm is made of low activity lead, the two inner centimeters consist of 400-year-old , very low activity lead. Figure 1 above gives a drawing of the lead-castle with detector and probe. 0.5 cm of very low-activity copper reduce the induced X-rays originating from the lead. The window between detector and probe is 0.07 cm thick and 6 cm in diameter. An intrinsic Ge-detector (semiplanar) of about 5 cm diameter and about 1 cm thickness reduces the volume of the detector as far as possible. Since most gamma-rays are in the energy region of 30 - 60 keV, the probe is placed in a plastic box of 5.9 cm diameter and only 1.7 cm height to account for the selfabsorption in the sample.

3. Background

The figure below shows the background spectrum accumulated over 32.8 days in the region of 0 - 256 keV with about 31 channels per keV. The only gamma-rays which can be seen are the X-rays of lead from 72.8 keV to 87.3 keV. The background rate is about 200 counts per keV and 500000 seconds in the region of 30 - 60 keV resulting in an average background rate of about 250 counts per keV and 500000 seconds.

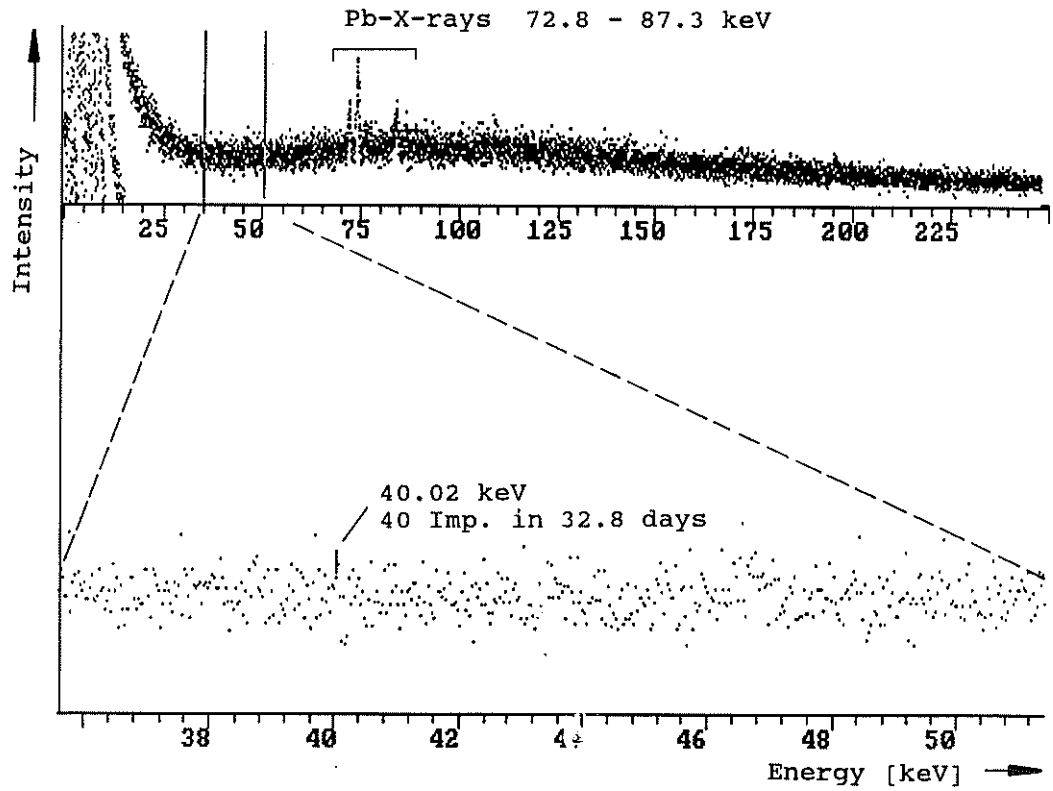

Figure 2: Spectrum of the background accumulated over 32.8 days using 32 channels per keV

4. Efficiency

The detector efficiency is determined by using calibrated samples from PTB with the geometries filter, 20 cm^3 water and 20 cm^3 sand (SiO_2). The aluminum window should dominate the absorption below 20 keV but it shows that absorption starts noticeably below 40 keV. Between 40 keV and 120 keV the efficiency is greatest at about 23% for the filter geometry and about 12% for the water and sand box, then dropping at higher energies.

5. Measurement

The figure below shows a water probe (about 5 liters) reduced to 20 ml and measured over 3.1 days. The peak of Am 241 at 59.5 keV is clearly seen and leads to 1.7×10^{-2} Bq per probe or to 3.7×10^{-3} Bq/liter. The width at half maximum is slightly higher than 570 eV at 122 keV, so a resolution of the peaks between 30 and 60 keV in the order of 0.5 keV is possible.

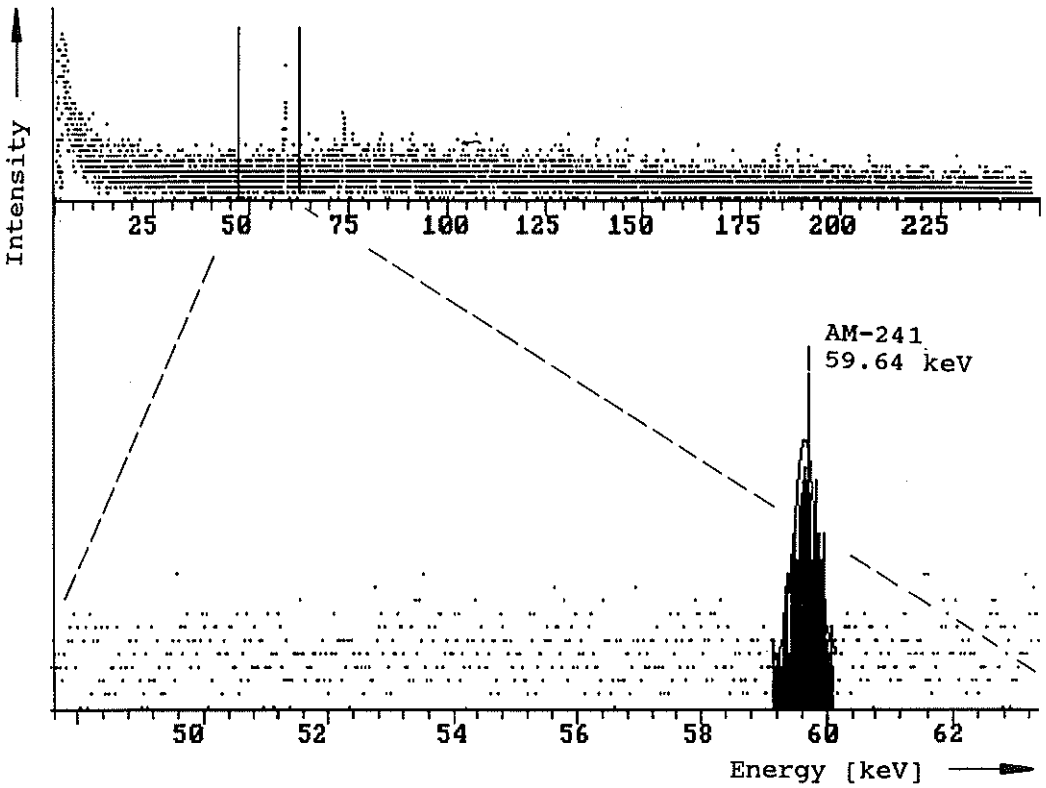

Figure 3: Spectrum of a 20 cm^3 waterprobe - measured over 3.1 days yielding 3.7×10^{-3} Bq Am-241 per liter

Table 1

Comparison of the lower limit of detection and the demands of the radiological protection ordinance using 20 cm^3 of material and 500 000 sec measuring time

Nuclide	Gammadata		Lower Limit of Detection (KTA 1503.1) [Bq]			10^{-4} of the Limit /1/in 20g Probe [Bq]
	Energy [keV]	Emiss.-probab.	Back-ground	Water [20g]	Sand [20cm^3]	
Th-228	84,4	1,22E-2	0,040	0,044	0,17	10
Th-230	67,7	3,76E-3	0,14	0,15	0,57	10
Th-232	63,8[1)]	2,70E-3	0,20	0,21	0,81	100
U-232	57,8	2,00E-3	0,23	0,31	0,88	10
U-233	54,7	1,82E-4	2,6	3,4	9,7	10
U-234	53,2	1,23E-3	0,39	0,52	1,5	10
U-235	163,3	5,08E-2	0,013	0,014	0,038	10000
U-236	49,4[2)]	7,8 E-4	0,65	0,86	2,5	100
U-238	49,5[3)]	6,4 E-4	0,79	1,0	3,0	10000
Pu-236	47,6	6,9 E-4	0,74	0,98	2,8	10
Pu-238	43,5	3,95E-4	1,4	1,8	5,3	10
Pu-239	51,6	2,7 E-4	1,9	2,5	7,1	10
Pu-240	45,2	4,47E-4	1,2	1,6	4,5	10
Pu-241[4)]	-----	-----	2,5	3,3	9,4	10
Am-241	59,5	3,59E-1	0,0013	0,0017	0,0049	10
Pu-242	44,9	3,73E-4	1,4	1,9	5,3	10

1) -If Th-234 (63,3keV) is present, the gammaline of 140,9keV has to be used for determination.

2) -If U-238 (49,5keV) is present, the gammaline of 112,8keV has to be used for determination.

3) -If U-236 (49,4keV) is present, the gammaline of 112,8keV has to be used additionally for determination.

4) -using the daugther-nuclide Am-241 after a build-up time of 4 months

6. Lower limits of detection

The table above shows the lower limits of detection for the main nuclides concerned in dependence from different materials. It is evident that material containing no activity at all (background) result in very low limits of detection. Compared to the water measured above, the limits are slightly higher. A sand matrix containing about 6 Bq/g of Uranium simulates a realistic probe with a rather high inventory of alpha-emitting nuclides. The lower limits of detection are determined according to the technical rule KTA 1503.1 /2/. To convert these limits in the new standard DIN 25482.5 /3/, the values have to be multiplied roughly with the factor 1.5.

7. Result

In the table above, the last column shows the maximum admissible contamination per probe (20g) derived from the 10^{-4} of the exemption levels of the radiological protection ordinance (/1/ §2, 4 and 83). A comparison of the lower limits of detection with the maximum admissible contamination shows that in most cases the observance of the latter can be done with this type of measurement. In special cases information about the mixture of nuclides, and the amount of well-detectable nuclides (U-235 and Am-241) as well as an information on the build-up time of daughter nuclides is necessary to achieve a final statement.

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ENVIRONMENTAL SURVEILLANCE OF THE *ASSE SALT MINE* AS A MODEL FOR THE SURVEILLANCE OF A FUTURE FINAL REPOSITORY FOR RADIOACTIVE WASTES

Ingo Müller-Lyda, Herbert Meyer, Rolf Stippler
GSF-Forschungszentrum für Umwelt und Gesundheit, GmbH
Institut für Tieflagerung
Theodor-Heuss-Str. 4
D-W 3300 Braunschweig, Fed. Rep. of Germany

Summary

The surroundings of the Asse salt mine are subjected to routine surveillance measures according to a basic immission measurement programme. The measurement results clearly indicate that no long-term changes of the radiological situation at the site have taken place due to the air exhausted by the mine, which was to be expected in view of the prevailing emission and propagation conditions. Thus, on the one hand, the legal and administrative stipulations with regard to the radiological immission surveillance are adhered to, and, on the other hand, important data for the validation and for the public information are provided.

1. Introduction

Since 1966 the Asse Salt Mine is being used by the GSF for R&D-activities concerning the development of methods for the final disposal of radioactive wastes. In the course of these works a total number of 125.000 canisters containing low-level radioactive waste and 1300 canisters filled with medium-level radioactive waste were emplaced into empty excavation chambers of the salt mine. Thereby the mine personnel, the mine itself and its surroundings were surveyed for radioprotection purposes according to legal stipulations and those of the Mining Authorities. Moreover, ever since 1966 a variety of samples were also taken and evaluated from the surroundings of the Asse salt mine in order to determine the environmental radioactivity. The environmental surveillance measuring programme has been set up in accordance with the "Guideline on Emission and Immisson Surveillance of Nuclear Plants". It was continuously adapted to the current requirements and coordinated with the Mining Authorities. This survey covers possible changes of the activity contents in water, air, and soil and the local dose. The monitoring performed within the scope of this programme indicated no measurable influence attributable to mine operation. This result is in good agreement with the potential radiation impact due to mine operation, as calculated on the basis of radioactive emissions.

2. Objectives of Environmental Surveillance

Apart from those radionuclides which are actually emitted with the mine air, such as Radon 222 and its progeny, Tritium and C 14, the measurement programme for environmental surveillance is concerned with the pathways for the propagation of radionuclides via air and water which must principally be taken into consideration in connection with the disposal of radioactive wastes. As these radionuclides are generally also present in the uninfluenced natural environment, the emission surveillance has to focus on the detection of slowly progressing changes of the radiological situation. Moreover, in view of the special situation of an R/D-facility as regards the methods of radioactive waste disposal it follows that the surveillance measurements are partly performed as accompanying scientific investigations and therefore exceed by far the extent required for a mere environmental surveillance. Particularly the examination of ground- and potable water represents a merely precautionary measure since there is no contact whatsoever with the emplaced radioactive wastes. The geological and hydrogeological conditions are being investigated in a separate research project and are considered within the localization of the sampling points. Thus the following objectives are to be pursued by the emission surveillance measurements performed in the Asse salt mine:

Tab. 1: Operator's Immission-Surveillance Measurement Programme in the Environment of the Asse Mine

Surveyed Medium	Parameter	Required Detection Limit	Sampling rsp. Location	Method and Frequency
Air				
- γ-radiation	dose rate	10 nSv/h	8 locations, one always in exhaust air direction	4 short-term measurements at various locations, one always in exhaust air direction
	local dose	0,4 mSv	10 TLD at fence, 6 at the railroad track, 24 in 1-2 km circles	semiannual evaluation
- Aerosols	short-lived total-α-ß-concentration			continuous registration
	total concentration of long-lived		location of maximum impact in most frequent exhaust air direction, and one reference	
	α-activity	20 μBq/m^3		
	ß-activity	40 μBq/m^3		
				continuous sampling biweekly evaluation
	single nuclide concentration, γ-spectrometry from .04 to 1.5 MeV	100 μBq/m^3		
	short and long-lived total α/ß-concentration	2 mBq/m^3	8 locations, one always in wind direction	monthly 4 grab samples at various locations one always in wind direction
Soil and Overgrow				
- soil	single nuclide concentration from .04 to 1.5 MeV	0,4 Bq/kg	3 locations, + 1 reference	2 grab samples annualy
	total ß-activity-surface concentration	4 kBq/m^2	3 locations at fence + 1 in wind direction	2 short-term measurements annualy
- gras	activity concentr. γ-spectrometry from .04 to 1.5 MeV	0,8 Bq/kg	3 locations, + 1 reference	2 grab samples annualy
Water				
- ground and surface water	total α-activity concentration	0,2 Bq/l	27 locations at sources, wells, and waste water channels	grab samples every 3-months
- potable water	single nuclides:		5 potable water supplies of nearby villages	monthly grab samples, semiannual evaluation
	Cs 137	2 mBq/l		
	Sr 90	2 mBq/l		
	Pu 239	0,2 mBq/l		

- Performance of an immission survey corresponding to that of a nuclear plant and in accordance with the stipulations of the Nuclear Regulations Authority,
- the detection of long-term changes of the radiological situation at the site,
- the acquisition of data in support of the hydrogeological research programme,
- holding in readiness of trained personnel and technical equipment for routine operations and assumed incidents,
- verification procedure and information of the public.

The results of the environmental surveillance measurements are published annually in the form of GSF-Reports [3] and serve to inform the local political bodies, authorities, and the press.

3. Measurement Programme and Results

The measurements are carried out by the mine operator in accordance with a programme set up by the supervisory authority. The programme is supplemented by measurements of independent institutions. The surveillance covers the media air, soil, water (cf. Table 1) and comprises a total of 368 annual measurements.

3.1 Local dose

The local dose is monitored by means of solid-state thermoluminescence dosemeters (TLD) positioned at the fence encircling the mine at 1 and 2 km distance and along the railroad track connection of the mine. This is to register the natural radiation exposure at the site, including the bandwidths of its local and temporary fluctuations. A comparison of the measurement data from two positions (Fig. 1), one, for instance, at the entrance gate (guard-house), through which the transport vehicles bearing radioactive wastes passed up until 1978, and the second one at the fence in the most unlikely expansion direction of the exhaust air demonstrates that there are no significant discrepancies between the two measuring points. The temporary dose increase in 1986 subsequent to the Chernobyl incident stands out remarkably.

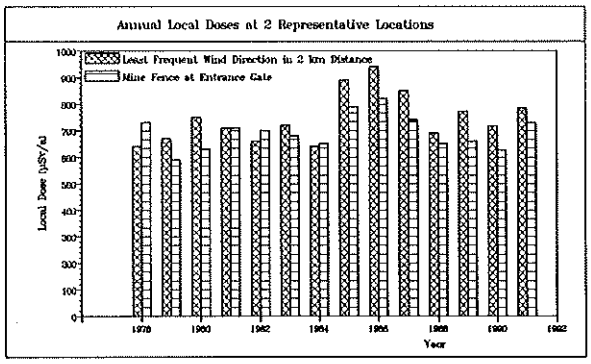

Figure 1

3.2 Aerosol contents of the surrounding air

The aerosol activity of the surrounding air is being measured at two locations by means of filter aerosol-monitors. One measurement point is situated at the mine fence in the most frequent wind direction at approx. 300 m distance from the mine diffusor and another one at an even greater distance serving as reference. With these monitors the short-lived α/β-aerosol activity of the radon-222-progeny is measured. After a sampling period of two weeks the filters are first analyzed for the long-lived α/β-activity and, subsequently, for single radionuclides using γ-

spectroscopy. The representation in Fig. 2 shows that the radon progeny detected at the immission measurement location is not attributable to the emission from the mine. In other words, if the wind blows in direction of the measurement point (sector WNW), then the radon concentration is comparatively low. Since, furthermore, the highest measurement values occur at low wind velocities, the radon can be attributed to emanation from the soil. Correspondingly this also applies to Pb 210 (Fig. 3). The concentrations measured in the exhaust air and in the surroundings by means of γ-spectroscopy both show uncorrelated fluctuations, that is the concentrations at the immission measurement point obviously remains uninfluenced by the exhaust air from the mine.

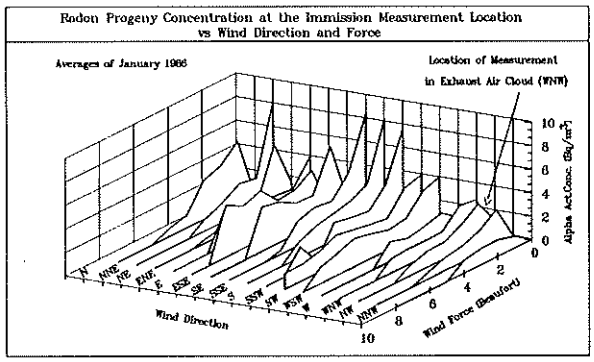

Figure 2

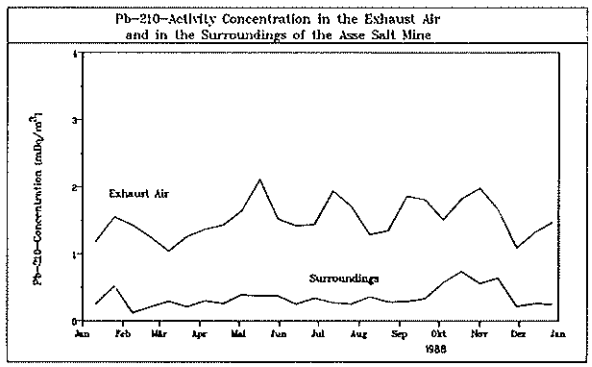

Figure 3

3.3 Soil and overgrow

Soil and overgrow samples are taken regularly from the nearer surroundings of the Asse salt mine and their activity concentration is measured. The sampling locations have been selected such that, corresponding to the most frequent wind directions at the site, the possible depositions of dust from the exhaust air are also taken into consideration. Three of the four sampling locations are situated directly at the mine fence while another one lies approx. 2 km in southwestern direction of the mine serving as a reference.

Sampling is carried out twice a year. Following a suitable preparation the samples are filled into a 1-l-Marinelli-beaker and measured for 20 hours in a γ-spectrometer. Here a detection limit of 0.6 (overgrow), resp. 0.3 Bq/kg (soil) can be achieved. As an exemplification the annual averages of the Pb-210-contents of the samples taken from 1982 to 1992 at the point of maximum radiological impact and of the reference location are compared. This comparison shows that no influence is exerted by the emissions from the mine (Fig. 4).

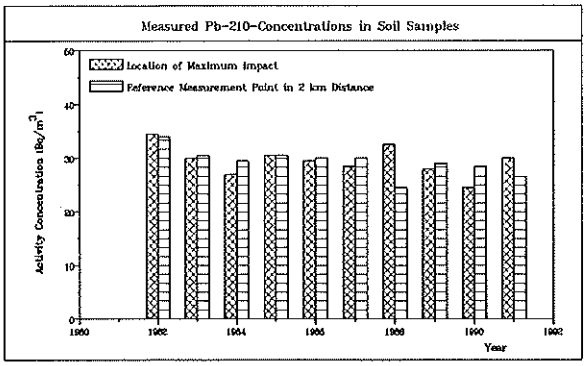

Figure 4

3.4 Ground, surface and drinking water

As there is absolutely no contact between the ground water and the emplaced wastes and since no operational sewage is released from the mine, surveillance of the water represents a merely precautionary and verfication measure. However, the results of these measurements are used within a hydrogeological research programme which is being prepared at the Asse site. For instance, in coordination with this project 1-l ground and surface water samples are taken four times per year from selected locations and examined for their ß-activity. Here the total ß-activity from the residues of the boiled-down samples is measured on a large-area counter. Additionally, the potassium contents of the water specimen are measured by means of flame-photometry. The residual ß-activity is obtained after subtraction of the K-40-activity from the total ß-activity.

Considerable uncertainties are attributed to the aforementioned method where water samples of a high potassium content are concerned. That is why highly saline samples as well as all samples with a residual activity surpassing 0.4 Bq/l are additionally analyzed by means of τ-spectroscopy for individual nuclides. Here a detection limit of 0,1 Bq/l (referring to Co 60) is achievable.

The drinking water of the surrounding villages is, providing it originates from the Asse region, examined at regular intervals. Aliquote random samples are taken from the drinking water supply system every month, carrier solutions are added and they are mixed semi-annually into a 50 l sample. The analyses for Pu 239, Cs 137 and Sr 90 are performed applying routine-procedures by the GSF-Institute for Nuclear Protection (ISS). The detection limits are 0,19 mBq/l for Pu 239 as well as 1,9 mBq/l for Cs 137 and Sr 90, resp. Occasionally detected Sr 90 and Cs 137 indicate that in these cases ground water from the vicinity of the surface has been included. Due to the fallout from earlier above ground nuclear weapons tests and the Chernobyl incident, Sr-90 and Cs-137-concentrations are frequently encountered in these surface-near waters.

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METHODOLOGY FOR EVALUATING THE RADIOLOGICAL IMPACT OF RADIUM-RICH BY-PRODUCTS STORAGE SITES

M.C. Robé, V. Labed, J. Le Bronec, F. Goutelard, J.M. Maurel
Institut de Protection et de Sûreté Nucléaire, Département de Protection de l'Environnement et des Installations, Service d'Etudes et de Recherches en Aérocontamination et en Confinement - IPSN/CEA, 91191 Gif-sur-Yvette Cedex, France

Summary

The radiological impact of ^{222}Rn emanating from radium rich by-product storage sites (i.e. uranium ore and ore tailings storage sites at processing installations, storage centres, etc.) can be evaluated by:

- site investigations including the following:
  . characterization of the source term by measuring the specific ^{222}Rn activity in the soil of the site, the ^{222}Rn exhalation rate of the site, and the external dose rate.
  . measurement of specific ^{222}Rn in the air above the ground and of the specific potential alpha energy of short-lived radon daughter products

- a similar investigation in the near and/or distant environment of the site.

The investigations are supplemented by relatively long-term radiological monitoring of the environment (measurement of potential alpha energy at several stations using a procedure to be determined in the light of the results obtained from the above).

Introduction

In all countries that process or have processed uranium ore, there are storage sites for radium-rich materials that may have a radiological impact on the environment.

This involves uranium processing plants where ore is stored before processing and sometimes also ore residues resulting from this processing. Waste produced from mining and processing associated with the industry of rare earth elements or phosphates are also stored at some storage centres.

Among the radioelements contained in these materials, radium 226 and therefore radon 222 are generally found.

Of course, these products are of natural origin, but they still require precautions to be taken in order not to significantly increase risks for present and future populations.

Below, we will describe a methodology for evaluating the radiological impact of such storage sites with respect to the radon emanating from the radium-rich products.

1. Evaluation methodology

The initial objective of the evaluation is to characterize the site and its more distant environment ("regional background"), and then specify a relatively long-term radiological monitoring protocol.

This evaluation is based on a set of measurements and requires several days.

The site and its environment are characterized in three main phases (fig. 1):

- characterization of the source-term [1],
- study of transfer in the environment,
- measurement of the regional background [2], [3], [4].

. Phase 1 :
Characterization of the source term mainly includes measurements of ^{222}Rn emission flux at ground level and at specific points.

Measurements are made according to an accumulation techniques which consists in determining, at a given time, the activity concentration of radon collected in a container of which one open side is applied against the ground. Using the device requires a cleared ground area of at least one square meter.

To determine the activity concentration of the radon, two air samples are collected in scintillation flasks in which a vacuum was previously created.

For each flux measurement point, once the two air samples in the scintillation flasks have been taken, a sample of the soil under the container is taken to determine the soil's moisture content, at the time of testing. This is because radon emission flux varies with soil humidity. This is why flux measurements cannot be made during rainy or snowy weather.

The number of measuring points is determined according to the storage site surface area, being kept in mind that the investigation is continued if necessary on the basis of results.

This characterization of the source term may be supplemented by spot measurements of ^{222}Rn activity concentration in the soil and ^{226}Ra tests of material samples.

. Phase 2 :
The study of ^{222}Rn transfer in the environment is conducted simultaneously using a mobile laboratory. It is equipped with the following:

- an electrical power generation system which provides 24 hours of measurement autonomy,
- a navigation system,
- a data acquisition and processing unit,
- ionisation chambers for continuous measurement of the gamma dose rate, at approximately 2 m off the ground, and of the radon 222 activity concentration. The air samples are taken at approximately 1.5 m off the ground.

The mobile laboratory is used to measure the temporal and/or spatial evolution of a pollutant on times scales ranging from a few minutes to several hours, and distance scales ranging from a hundred metres to tens of kilometres. Several measurement procedures have thus been developed.

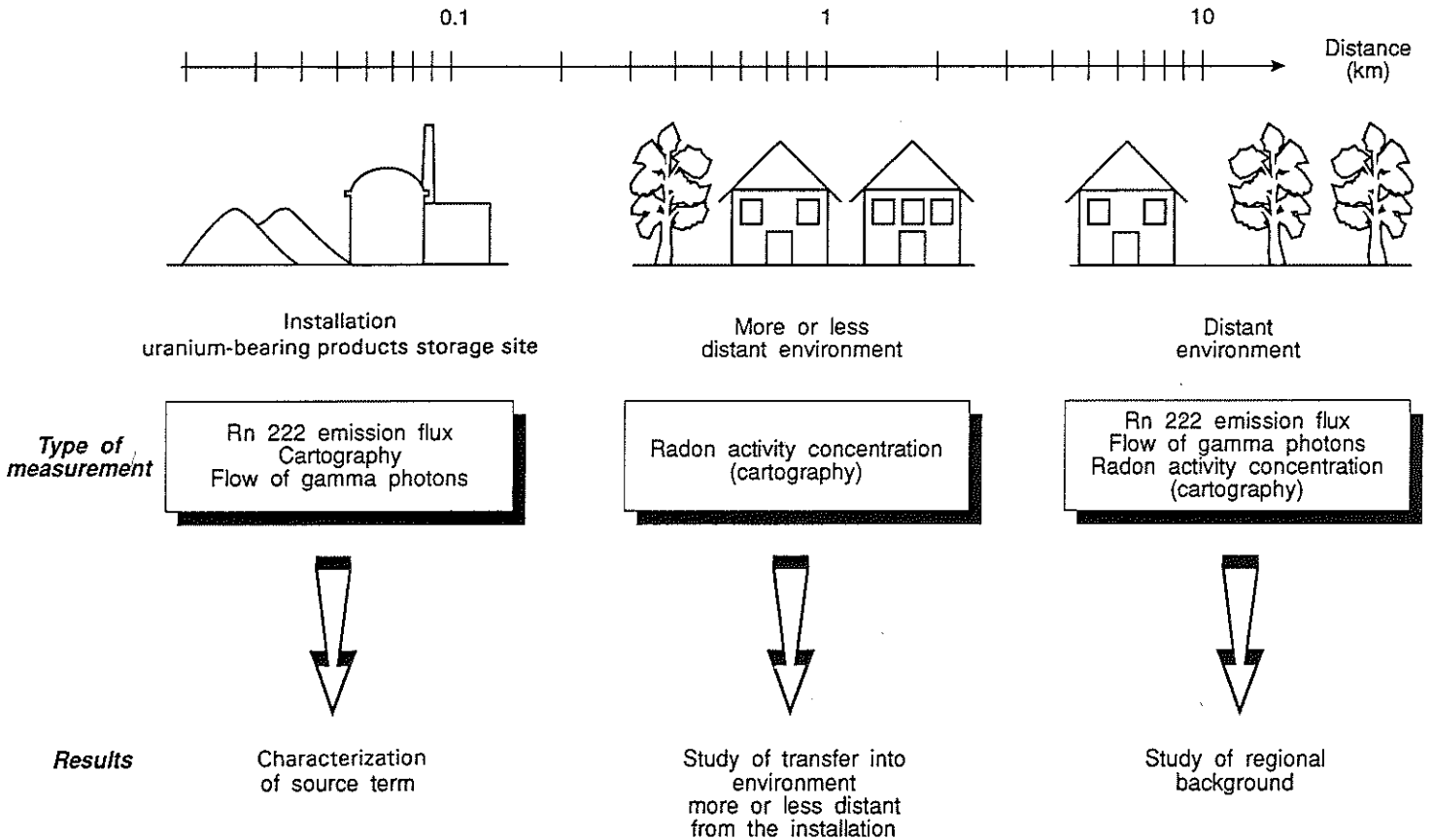

Figure 1 - The different phases for evaluating impact of a radium-rich by-products storage site

Measurements are made as follows:

- at fixed stations, at several points of the site, over periods of several hours to several days and under various conditions of atmospheric diffusion. This reveals variations according to time of the day which are related, for example, to temperature changes at night;
- along various points of paths determined according to the site's characteristics (i.e. road infrastructure, housing, orography, etc.) and according to weather conditions under various conditions of atmospheric diffusion. This reveals spatial variations;
- at semi-fixed stations : the mobile laboratory is immobilized for 4 to 5 minutes at a point of known coordinates (x, y and z) then moved by a few hundred metres (in accordance with the orography) to carry out new measurements.

This methodology is also applied in the storage site near-field environment. Some ^{222}Rn flux emission measurements are made, if necessary, in order to determine if the ^{222}Rn observed in the atmosphere is local in origin or if it is due to transfer from the radium-rich product storage site.

. Phase 3 :
In the environment more distant from the storage site (several kilometres away, outside of the site's range of influence but in an area with comparable geological characteristics) this same methodology is used in order to determine the ranges of natural ^{222}Rn emission flux levels and activity concentrations within the region.

The limits of the storage site's zone of influence may thus be determined with respect to these reference levels.

2. Relatively long-term radiological monitoring

On the basis of this evaluation, radiological monitoring may be efficiently carried out to assess the potential additional exposure for people living or working within this zone of influence (the "critical group"). It consists of measurements of the specific Potential Alpha Energy (sPAE) of the radon short-lived decay products, integrated over approximately one month, and representative of the "critical group" and the natural level [3], [4].

3. Conclusion

The evaluation methodology described can very rapidly provide information useful at various levels, including for:

- the site operator,
- the authorities,
- the population.

Firstly, it makes it possible to determine the radiological characteristics of the radium-rich product storage site and to determine its zone of influence in the environment, while taking the regional background noise into account.

Secondly, if critical population groups are identified, this evaluation makes it possible to propose an efficient monitoring network installation for the relatively long-term.

Lastly, it constitutes a database for assisting storage site management. It should be noted that this evaluation methodology may be applied to a site that is being operated, that is no longer operated, or that has been already restored.

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RESULTS OF ACTIVITY CONCENTRATION MEASUREMENTS OF SAMPLES FROM AN URANIUM MINE AND ORE-PROCESSING PLANT

K. Prokert, G. Mann
Institute of Radiation Protection Physics, Physics Department
Dresden University of Technology,
Mommsenstraße 13, Dresden, O-8027, F.R.G.

Summary
Using gamma spectrometric measurements and X-ray fluorescence analysis of typical samples from the territory and the production process of the Königstein uranium mine were investigated. The results show that ore residues as well as the solid waste from the chemical process and the ground inside the plant have values of specific activity which are higher than the "BfS-limitation" of 0,2 Bq/g.

1. The Territory and the Production Procedure

Near Königstein (Saxon Alps) an uranium mining and ore processing plant of the WISMUT AG has been in operation since 25 years. In comparison with other complexes of this industry in Saxonia (e.g. districts Zwickau, Schlema, Aue) or Thuringia (district Ronneburg) it is an only small exploitation plant. However, this object is very interesting and important for studies of the impact of such a special production on the environment, because another uranium exploitation technology is used on this place.

The uranium deposit occupies an area of only 25 km^2 and lies in a depth of 180 ...250 m under the ground. Due to the special geological conditions the stratum of poor uranium ore (U-content appr. 650 ppm) is completely isolated from ground water strata for the winning of drinking-water. Therefore an underground solution technology using following procedure is in operation. Starting from a system of tunnels the ore stratum will be crushed by breaking, then the leaching takes place using dilute sulfuric acid (molarity 0,03). The acid

streams slowly through the crushed material and yields the "rich solution" by dissolution of the uranium. This product is pumped onto the surface and transported into the chemical part of the plant. From this "rich solution" the uranium is separated by an ion exchange process, the separation of other elements follows by a precipitation step. After sedimentation of the mull the cleared and acidified "poor solution" returns into the solution process of the uranium ore. A small part of crushed ore which was transported to special places in the plant also yielded a uranium containing solution by the same leaching process. It is added to the "poor solution". In this manner a closed cycle of the liquid phase is realized. The solid residue the so-called "filter cake" is the waste of the procedure and must be deposited.
During the process of shut down of the plant, it is now going on, the circulation process of solutions cannot be stopped at once. It must be continued under changing conditions (slowly decreasing of the sulfuric acid content) until the content of uranium and acid attains values which permit a stop of the circulation without significant pollution of ground and water in this and other strata.

2. Measuring Methods and Materials

The specific activity of the various samples was determined by high resolution gamma spectroscopy. For a few samples the uranium content was determined by X-ray fluorescence spectroscopy with polarized X-radiation.

2.1. Gamma Spectroscopy

Ge(Li)-detector (CSFR) with preamplifier (ORTEC, Typ 20-3F) and amplifier (ORTEC, Typ 572).
Rel. efficiency: 4,6 %, energy resolution 6 keV, both values are related to a gamma-energy of 1332 keV.
Registration by an ORTEC-MCA-card (4096 chanels) in a PC using the program MAESTRO.

2.2. X-Ray Fluorescence Analysis

"SPECTRO X-Lab" (SPECTRO X-RAY INSTRUMENTS) with Rh-Anode,

B_4C-polarizator, LN_2-cooled 30 mm^3 Si(Li)-detector (Be-window, energy resolution 155 eV for Mn-k_α-radiation).
The integrated software of the equipment permits a qualitative and quantitative determination of elements with atomic numbers Z > 10.

2.3. Sample types
The following samples taken from the production procedure and from the territorium of the plant were measured:
- Dead rocks (sandstone), untreated and leached ore,
- Filter cake,
- "Rich solution" and "poor solution",
- Water from the stratum with uranium ore,
- Water from the bottom of a dump,
- Ground sample from a road in the territory of the plant,
- Ground sample under vegetation from the territory of the plant,
- Ground sample under vegetation outside of the plant.

3. Evaluation and Results
The evaluation of the measured spectra permits the determination of elements and nuclides as well as their concentration or their specific activity.
The evaluations are based on relative methods, i.e. that a calibration of the equipments with standard samples must be undertaken. Using commercial programs or a program [2] developed in our institute the influence of specific sample properties (density, volume, attenuation coefficient) on the measured values was taken into account.
The Fig.1 and Fig.2 show typical gamma spectra of various samples. In Tab.1 are given values for specific activities and for uranium concentrations.

6. Discussion
The measured values of specific activities of various samples and the limitation given from BfS [1] with 0,2 Bq/g show on one hand the necessity of special treatments for wastes and ore residues. On the other hand we can conclude that the

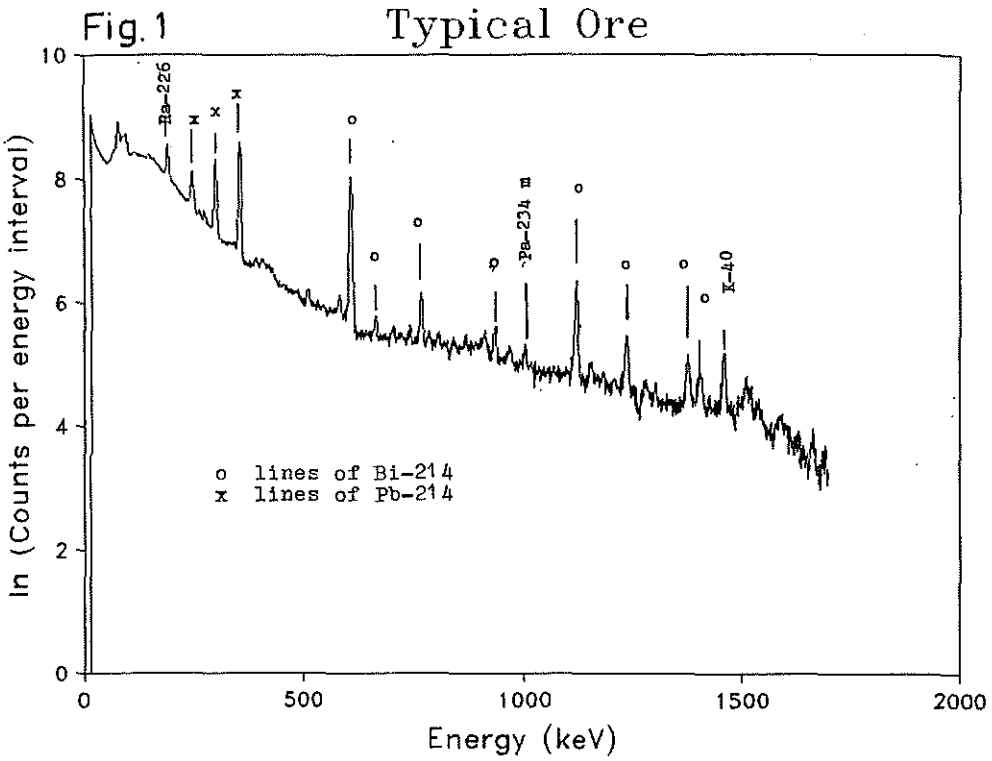
Fig. 1
Typical Ore
ln (Counts per energy interval)
Energy (keV)
Ra-226
Pb-234 m
K-40
o lines of Bi-214
x lines of Pb-214

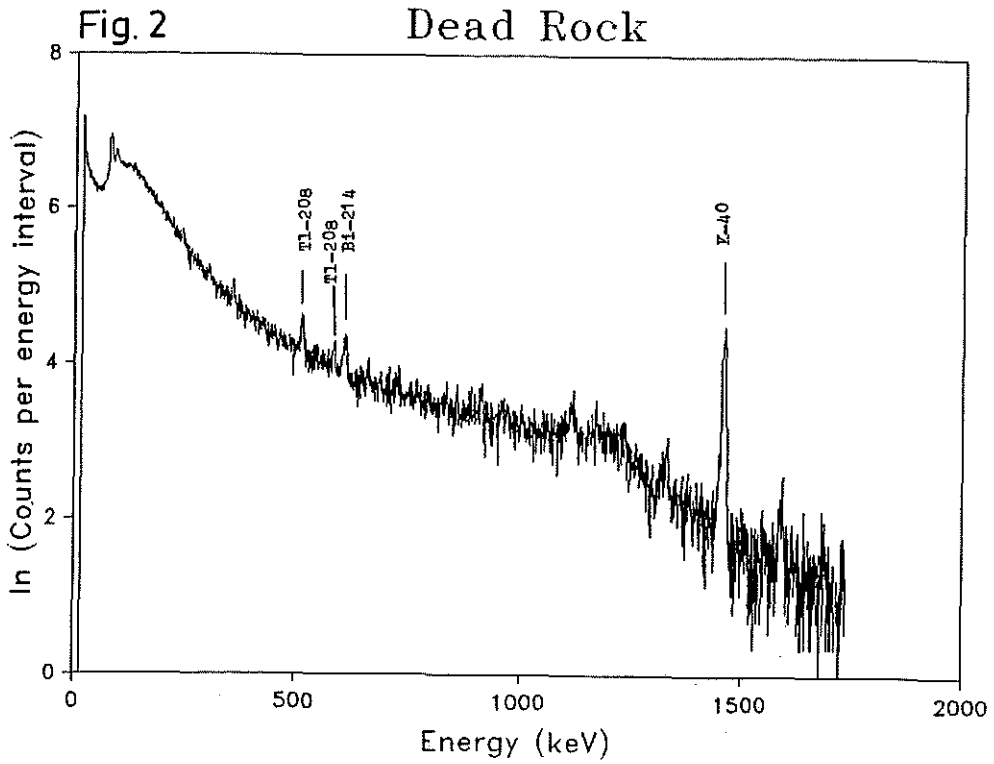
Fig. 2
Dead Rock
ln (Counts per energy interval)
Energy (keV)
Tl-208
Tl-208
Bi-214
K-40

Table 1: Specific activity and uranium content of samples

Sample	Specific Activity calculated for Ra-226 Bq/g	Uranium concentr. ppm
Dead rock (sandstone)	0,01 +/- 0,01	
Ore	0.79 +/- 0,05	200 +/- 4
Leached ore	1,71 +/- 0,1	180 +/- 5
Filter cake	4,43 +/- 0,3	240 +/- 4
Ground sample (road)	0,51 +/- 0,02	
Ground sample (inside)	0,94 +/- 0,02	
Ground sample (outside)	0,01 +/- 0,01	
"Rich solution" 1992,july	0,05 +/- 0,02	25 +/- 2
"Poor solution" "	0,04 +/- 0,02	< 2
Water (ore stratum) "	0,26 +/- 0,05	< 2
Water (dump) "	0,05 +/- 0,02	20 +/- 2

deposited dead rocks and the ground outside the plant have specific activities camparable with that measured in regions without uranium mining industry. Due to the chemical treatments during the production process the uranium content and the specific activity of samples do not show a correlation. This is caused by the different chemical behaviour of uranium and radium. It seems that the radioactivity of the residues of this industry is not the main problem here. The content of acid and weakly soluble sulphates in the stratum in the underground and in materials deposited in the dumps could give more problems in the environment.

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PORTALMONITOR FOR GOODS AND CONTAMINATION MEASUREMENT

Robert Baumgartner, VICTOREEN GmbH, Munich, Germany
Hans-Albert Ozimek, VICTOREEN GmbH, Munich, Germany

SUMMARY :

The PORTALMONITOR, type RBO, is applicable for the measurements of goods and freights on any kind of vehicle as well as for contamination measurement and penetration control of the vehicle itself.

A lot of regulations for these kinds of surveillances are already existing or being under consideration for final release in the near future.

The modular arrangement of the hardware components and the software program offers the possibility to measure according to each customer's specific application.

By varying the number and the position of the detectors the sensitivity and efficiency of the entire system is adaptable according to customers requirements.

TECHNICAL DESCRIPTION:

A adequate number of NaJ(Tl)-detectors is installed in a mechanical frame e.g. in columns or a portal assembly. The detectors are placed inside of lead-collimators to lower the background.
The angle of the collimator depends on the geometrical circumstances of the measurement.

The signals supplied by the detectors are submitted to amplifier/discriminator modules for single channel evaluation, or via amplifier units to a multi-channel-analyzer (MCA).

The electronic modules are easy to maintain and the voltages and signals are to measure by front panel plugs. Also the discriminator levels are adjustable at the front panel.
The modular structure of the software program allows many applications according to the requirements of the measurement and the customer.

Due to the fact that scintillation detectors are used, a nuclide-spectrum can be measured.
Therefore, each energy (nuclide) can be identified, and the entire spectrum can be controlled.
In case of exceeding an adjusted threshold an alarm is audible and visible.

Also potential free outputs are provided for remote alarm and display panels.

Various read-outs are applicable e.g.:

- Bar-Graph
- Counter
- Spectrum with or without ROI's
- Real Assembly Scheme with Industrial PC

The electronic can control the dependent equipment parts as:

- Alarm panels and signal lamps
- Keep doors closed during too high radiation levels
- Motors to move the monitor or the vehicle
- Air conditioning system to keep the detector-temperature constant (if required)
- Printer, Recorder and Modem
- Bar-code Reader

The lower detection limit of the system depends on the circumstances of the measurement as :

- Background Level
- Geometrical Parameters
- Measuring Time
- Corresponding Energy (Nuclide)
- Number of Detectors used

EXAMPLES FOR THE APPLICATION OF THE PORTALMONITOR:

The PORTALMONITOR is applicable for the measurement of the entire freight on trucks or waggons, or the contamination monitoring of any vehicle entering or leaving an area that has to be surveyed, e.g. a nuclear facility. Also a simple penetration check, whether nuclear material is transported or not, is possible with this system.

REFERENCE:

Since March 1991 a PORTALMONITOR is in use in Sollenau,AUSTRIA for the measurement of imported material from an Eastern Block country.
The system is fully-automated and self-powered and moves step-by-step along standing waggons or trucks, scanning segment for segment.

This PORTALMONITOR, that contains 10 detectors, has a lower-detection-limit of : 25 Bq/kg for Cs-137 within 15 min.

EXAMPLE OF A PRINT-OUT FOR DOCUMENTATION:

```
**********************************************************************
26. 3.1991  21:34  Wagen-Nr.: 57127596  Masse: 20230 kg  Länge: 13.33 m
Meßstellen: 20  Meßraster: 600 mm  Meßzeit: 900 s  Startposition:  2.81 m
Meßoffset nach Wagenanfang: 0.80 m  Messende vor Wagenhinterkante: 0.60 m
----------------------------------------------------------------------
     !    Meßgeber 1 bis 9, die Angabe der Werte erfolgt in Impulsen
 Nr  !    1     2     3     4     5     6     7     8     9     Bq/kg
----------------------------------------------------------------------
   1     919   681   894  1015  1265   959   972  1018  1665      20 Bq/kg
   2     985   640   931  1073  1351  1052  1038  1101  1708      22 Bq/kg
----------------------------------------------------------------------
Wagenwert:  21.03 Bq/kg
26. 3.1991  22: 5  Ende des Wagens bei :  4.68 m
***  Messung wurde abgebrochen !
----------------------------------------------------------------------
```

Part 4: Accident Management

ASSESSMENT OF ARRANGEMENTS SET UP IN FRANCE TO DEAL WITH NUCLEAR AND RADIOLOGICAL EMERGENCIES

by Paul GINOT

Chargé de Mission at the Institut de Protection et de Sûreté Nucléaire, Centre d'Etudes de Fontenay aux Roses, BP6 92265, Fontenay aux Roses Cedex, technical adviser to the General Secretariat of the Inter-ministerial Committee for nuclear security.

Summary

A few figures illustrate the field covered by the organisation which has been put in place to deal with the daily or exceptional emergencies which occur in the nuclear industry in the premises of the users of radioactive materials and during the transportations. Some examples are then given of actual cases of intervention, exercises, and training methods. The assessment then considers the regulation and management of the interventions.

The conclusion shows that the broad lines of the arrangements made for the control of emergencies which may arise in the various sectors and those which could affect the large nuclear installations are pertinent.

In the other hand, the handling of economic and social disturbances which would result from the dissemination of radioactivity in the environment is still poorly mastered. In this regard, a major role should be given to the social and economic participants and to their customary interfaces to the administration, particularly at local level. The promotion of this organisation in the coming years will depend on the increased awareness of these participants, and on the ability which they and the specialists in the nuclear industry have to train each other.

1 - First Chapter : field of the emergency organisation

We will begin by looking at the field which is covered in France by the organisation which has been progressively established in order to deal with any nuclear or radiological emergencies. This includes making an inventory of the sources of radiation and also of the frequency and level of gravity of incidents or accidents to which these sources give rise.

1.1. The sources of radiation

a. The electro-nuclear program consists of 56 nuclear reactors, and the mines and processing plants set up to cater for the full combustion cycle, as well as the reactors and laboratories spread over ten or so research and production sites, representing almost 150 large fixed installations described as basic nuclear installations.

b. The use of radio-isotopes and radiation sources involves some 3500 users of radio-isotopes in industry, 240 medical centres with a further 1400 in medical research. In 1991, the sources known as "sealed", with intense radiation, were supplying 200 gammagraphy installations and 2 large industrial irradiators. In 1992, 175 cobalt sources and 178 accelerators are used for radiotherapy. 27 sealed sources are used for blood irradiation. 132 services of curie-therapy employe sources of smaller activities. Dealing with radiodiagnostic the "non sealed" sources of small activity and treated as consumable, amounted to 135,000 deliveries.

20,000 and 35,000 X-ray sources are used for medical and dental diagnosis purposes.

Four large particle accelerators are used in the fundamental physics field.

c. The **transportation** associated with this activity concerns an annual traffic amounting to some 300,000 packages. The regulation distinguishes the radioactive materials, the security of which is limited to the package, and the nuclear materials which, when considered sensitive, are accorded real-time supervision during transportation.

About 700 such supervised journeys took place in 1991, 500 of which concerned transportation of **irradiated fuels** from nuclear plants to La Hague or Marcoule re-processing centres.

Defence-related activity in France must also be taken into account, including 12 **naval propulsion reactors** and one land-based prototype, in addition to other installations in which any accidents can affect national territory - such as those **in space or in other countries.**

1.2.The frequency of incidents/accidents and their levels of seriousness

a. In the electro-nuclear program in 1991, 500 events affecting basic nuclear installations were declared to the **industrial safety authority** - 350 described as benign, 138 concerning operational faults, and 12 incidents which called for further examination. Since the beginning of the programme, two serious accidents - Saint Laurent 1980, and La Hague 1981 - have resulted in the accidental discharge of radioactive materials into the environment, though fortunately these were without consequence.

b. In the case of transportation, approximately ten incidents annually require intervention by a specialist team for the purpose of performing checks and declaring safe. One serious accident - a collision in the North Sea - occurred in 1984, though with no radiological consequences (see paragraph 3.a.).

c. Over the whole field of activities, for the period between September 1990 and September 1991, the **health protection authority** performed some 60 checks for contamination on individuals, and some 20 checks for exposure. It also offered an opinion concerning 200 events, most of which concerned nuclear sites, 199 proved to have no harmful consequences, but a severe accident occurred on an industrial accelerator. In this last case the workers were injured with severe burns, some requiring intensive medical cares.

d. For some 30 or so years, about forty serious **localised irradiation incidents** have been recorded in the medical field, two thirds of which were associated with X-ray work. In the industrial and research areas, 200 localised serious irradiation incidents occurred, 70 of which were X-ray related.

No serious **whole-body irradiation** incidents have been noted in France. On the other hand, French doctors and biologists have had to treat 18 such cases resulting from incidents abroad. To put the position in perspective, in the period since 1940 French doctors have recorded 80 **mortal irradiation accidents** throughout the world, but none in France.

2 - Second chapter : the management of the daily events

These results indicate that, in the nuclear and radiological area in France, operating anomalies or incidents are occurring on a daily basis. An organisation which can deal with these emergencies is therefore called upon frequently. The technical procedures set up by he operator, the services provided by hygiene and security system or a call to the **first-aid service provided by the fire brigade in France** (Sapeurs Pompiers), are generally adequate to cover most situations. However a supervision and alarm system is needed in order to raise the level of action if this is required, calling in more heavy-duty resources located at the level of the region or zone, or at national or international level.

3 - Third chapter : Accidents involving the national organisation

The following is a typical selection:

a) Saturday 25 August 1984 - collision of the Montlouis in the North Sea

The freighter Montlouis, with 30 containers of uranium hexafluoride on board, collided with a car ferry at a point 12 kilometres to the north of Ostend. There were no victims.

The subsequent assessment of the accident indicated that:

- the activation of the alarm system was special : the persons in charge of the safety of the installations and of the public health have been informed fast and they react immediately, giving a preliminary assessment of the situation. So the authorities have been informed soon. On the reverse the official answer to the mediatic request was slow. So for one or two days the newspapers reported the information brought by the association which are usually severe regarding nuclear energy. This delay is mostly due to the decision to have only one source of official information, of course for coherence concern.
- the preventive measures which had been taken in respect of the containers had been effective, and that their sealing had remained intact. Nevertheless certain technical recommendations were made.
- the action taken to rescue the passengers, to recover the containers, and to effect chemical and radiological monitoring, had been satisfactory.

b) 26 April 1986 - the accident at Chernobyl

This accident set off an intense level of activity in France under the stimulus of anomalies in radiation levels which had been detected at nuclear sites, certain universities, and by associations for the defence of nature, aided by information from abroad, by the activity of the press, socio-economic commentators and the farming community with mainly cattle breeders and distributors. It is remarkable that the strong personalities in the nuclear field reacted without reservation, each coping with the event to the limits of his competence, but that the national organisation has not been set up. This revealed that the French national organisation was in fact more oriented toward the nuclear plants themselves, and had little to offer in the off-site post-accident situation.

This accident had important consequences for the national organisation, and resulted in:

- increased attention to the post-accident phase : assuming in an operational way the European Community directives on the marketing of food stuff and the health recommendations of the international Commission for Protection against Radiations,
- the setting up of the means of intervention on a hostile environment,
- deployment of territorial radioactivity monitoring systems,
- an increased responsibility on the part of Members of Parliament. Since, the Parliamentary Bureau for the assessment of scientific and technical policy has been conducting in-depth enquiries into the key areas,
- a requirement for transparency of information to the public and the media,
- access to the experience acquired in the ex Soviet Union,
- an increased role for the international organisations.

The list of consequences could be lengthened.

c) Summer 1988 - the Cosmos 1900 satellite

This satellite, powered by a nuclear reactor, did not in fact fall to earth, but this possibility was feared over a period of several months. Such an event on a previous occasion had affected Northern Canada, resulting in slight contamination of 50,000 square kilometres, in addition to the deposit of scores of solid debris representing a serious danger in case of physical contact.

The national organisation was given several months to adapt itself to such a situation, and to set itself up. An assessment brought out the following main points:

. The space phase caused the inclusion of the Centre National d'Etudes Spatiales and its installations at Toulouse to be brought within the orbit of the national organisation. This centre had been very dependent on the NASA for the monitoring of satellites. The level of uncertainty regarding the point of impact remains very high up to the last few minutes.

. The study on safety and protection was based mainly on information from the impact of Cosmos 964 in Canada.

. Preparation of the terrestrial phase was directly controlled at the Prime Ministerial level. This called upon all of the mobile teams of the Direction de la Sécurité Civile, the Atomic Energy Commission, the COGEMA, and the territorial resources of the national police force, the Gendarmerie and the Sapeurs Pompiers (fire and emergency service).

. It has been decided to make a big effort in the monitoring of the ground activity by air born detection teams. Using civil service helicopters, the Direction des Applications Militaires of the Atomic Energy Commission set up the detection strategy and the Air Force was in charge of the air traffic and the logistic for helicopters. Taking into account the large area to be covered, it became clear that this operation would take a very significant amount of time, and that contributions from other sources of information would be very welcome.

. An exercise in terrestrial detection was mounted spontaneously for the benefit of the Sapeurs Pompiers teams. This was a very useful and much-appreciated event. Nothing of this kind has been mounted for aerial detection.

. The raising of doubts in the case of an actual fall of satellite debris, given the natural or artificial radiological irregularities over the terrain would have given rise to a major problem. If the geologists of the COGEMA had been called, they would doubtless have been taken somewhat unawares.

. Preparation of the media for the event was virtually non-existent, and in fact, in a real event, one might expect a multitude of false alarms and rumours.

. Cross-border relations were practically non existent. It is obvious that this would have been an important factor in an actual case.

d) 14/15 June 1990 - the Jacques Coeur interministerial exercise

This operation was effected by the Belleville power station and the Préfecture du Cher in Bourges, involving its own services, together with the decision, coordination and expertise centres in Paris.

17 contracted professional journalists worked on the real-time production of material for the press, radio and television. 30 local people, including elected officials and commercial or residential representatives were invited, playing the roles which they would play in the case of an actual accident.

The exercise was limited to the exchange of messages, being conducted in two phases with independent scenarios, the first one concerning a serious accident involving the risk of core melt-down and the second the contamination of neigbouring agricultural zones over 200 kms.

The following information emerged from this exercise:

. The emergency organisation, which included the operator, the authorities and the experts, had functioned well, and had proved to be appropriate to the installation phase. Security measures had proved to be effective.

. The work of the journalists had been good, and the material produced had greatly influenced the exercise. The public relations of the Electricité de France had coped with the situation, and the position of this organisation had frequently succeeded in making itself felt. The press spokesman for the Préfecture and his assistants had received useful training, but they had not really offered a service which was up to the occasion.

. The simulation during the phase dealing with environmental contamination had not covered the associated themes successfully. This had hampered the players, and gave rise to a crisis within the crisis - interesting in retrospect.

. The central experts in health matters had coped well with the situation during the post-accident phase, and the Préfet has consulted it continuously. The people in the local services concerned with public safety, health matters and the food chain, performed useful training, but the services themselves and the overall organisation did not live up to expectations for an event of this scale.

. The elected representatives and local people, aroused by radio broadcasts and agency flashes, reacted violently to the simulated events, and made many demands on the administration. However, they received only scant response. Some of the commercial people, isolated in a similar manner, proved capable of acting alone.

e) 14/15/16/17 October 1991 - manoeuvres at Cadarache

This operation started at the post-accident stage and concentrated on agricultural production matters. Contamination of the ground was simulated for about 50 kilometres around. A chain of resources was made available, from contaminated earth to a control centre at the Préfecture, and including aerial detection, measurement on the ground, water and agricultural products, data processing, preparation of situation reports, recommendations, protection of personnel, etc.

The operation was controlled by the territorial authorities, the Atomic Energy Commission, the Compagnie Générale des Matières Nucléaires (COGEMA), and a mobile radiological intervention cell from the Sapeurs Pompiers.

The following were notable:

. The simulation and the scenario were successful. This was mostly set up by the players themselves, and drew largely on the lessons learned at the Jacques Coeur exercise.

. The intervention teams were very well received by farmers and stock farmers, who had been warned by the authorities beforehand.

. The contribution of veterinary and agricultural services was highly significant - general agricultural statistics, the cultural calendar, lists of sample producers and breeders, etc.

. The operation of a "contaminated zone entry/exit" station : this station was very greedy in terms of radioprotection staff.

. The strategy and tactics for intervention, coupled with the contribution by the Direction des Application Militaires proved very effective.

. The use of equipment by the mobile and laboratory teams for analysis purposes: the change from the daily work on a nuclear site to work off the site in an emergency was tested and evaluated.

. The local and central resources in expertise showed that the staff was competent but too few in number. Computer resources proved useful but require more work to make them operational.

4 - Fourth chapter : training

The activities mentioned above illustrate the manner in which the emergency organisation operates and develops. However, for a better understanding of this running-in process, account should also be taken of the following three initiatives which reinforce the **training** and **proficiency** of some participants who are particularly exposed or aware of the nature of their roles:

a) First witnesses and actors to the emergency, the operators in the control room at nuclear power stations have the advantage of continuous training, using the **operation simulators** at Paluel and Caen and at the Bugey training centre. By the use of a method which has already proved itself in the aeronautic field, they can become familiar with the type of emergency situations which can arise in a control room, and can therefore acquire the appropriate reflex, cool reaction, and intelligent response.

b) In the second line, and the source of vital information for the players downstream, the safety experts now also have the **SIPA1 and SIPA2 research simulators** at Villeurbanne and Fontenay aux Roses. These instruments make use of the most recent informatic codes and thermo-hydraulic data available and the power of a Cray computer. They set up an elaborate simulation of the progression of the main physical parameters of the reactor up to the point where the core is damaged, in situations which can be varied in real time. They are not designed to follow up actual events in real time. However the experts thus have the benefit of prior training in abnormal situations, which qualifies them to make diagnoses and forecasts concerning actual physical phenomena. All sorts of situation can be simulated, with the ability to question the computer, and based on its responses, it is possible to develop a personal response to any crisis, within the limits of the simulation.

c) At the third echelon, providing emergency protection for the population, the specialist Sapeurs Pompiers (fire and emergency services) must also receive individual training and practice. A **radiation-protection manual** has just been produced, and this constitutes not only an operational guide but also a reference, and an important step toward the assimilation of the radiological and nuclear culture by the civil security professionals.

d) At the fourth level, involving protection in the longer term and concern for the welfare of the economy, a large farmers' association has published a collection of **illustrates leaflets on the contamination of agricultural land.** These documents, which are the first of their kind in France, are aimed at helping the profession to assess any problems with which they are liable to be presented, and to prepare, on their own initiative, a suitable response to handle any disaster.

5 - Fifth chapter : management

5.1. The law and regulation

Among the publications which should be noted, there is a newcomer : The **Directive du Premier Ministre, SGSN 1444**, dated 1 July 1991, which give precision on the organisation of public authorities in the case of an accident affecting a civil nuclear installation. These instructions are the result of an assessment of the national crisis organisation, which was carried out, by the participants themselves, at the end of the Jacques Coeur exercise (see paragraph 3.d). **At present, the themes in this directive cover the post-accident phase and communications - collection and use of social and economic data, in particular concerning the food chain; assessment of exercises of local origin; learning from acquired experience; training of the préfectoral corps and teams, and the handling of information to the public and the media.**

5.2. Management of the intervention process

The cases of real emergency as described in section 3 show that the emergency organisation was adapted to suit each of these operations. Without doubt, this will be the case for all serious situations, if only due to the activity of the political authorities, for which planning is virtually impossible. It will at least be necessary to have a simplified presentation of the role of each of the participants and to draw up organisation charts for some typical situations, if only to establish a pattern for those operations which recur frequently, and for the training of emergency teams.

Five schemas deserve mention in this context:

a) The simplified **table of tasks covered by the central public authorities**, concerned in one way or another with radiological and nuclear emergencies (see figure 1). This list is capable of extension. The list covers those services which, up to now, have actually participated in the exercises, and the subsequent reflections and the planning.

b) The schema used by the Direction de la Sûreté des Installations Nucléaires for the phase of the accident concerning the installation itself. Figure 2 shows the arrangement made for the nuclear power plants of Electricité de France. It is put to test several times a year and has been well run in.

c) The schema used during the manoeuvre at Cadarache in 1991 (see paragraph 3 and figure 3). This schema is limited to the food chain ; it is of a simplified nature by virtue of the small number of participants and forms the basis of a chapter on **post-accident plans** covering the first days and weeks after the accident.

d) The schemas used by the Direction de la Sécurité Civile to show:

the operation of local resources of the fire and emergency services (Sapeurs Pompiers) which are available to all Mayors at city level and Préfets at Département or zone level, with mobile cells for radiological intervention, special military units, and all of the central resources at the disposal of the Direction (figure 4). The whole response organisation is coordinated by the **Centre Opérationnel de la Direction de la Sécurité Civile (CODISC)**. This centre operates continuously, dealing with all natural and technological risks, whether nuclear or not.

e) the increase, where required, in the level of mobile emergency resources sent into the field (figure 5).

6 - Sixth chapter : conclusions

We have been examining some of the main factors which, in the course of the last few years, have caracterised the actual operations and training activity, as well as in regulation, ideas and practices in the management area. Having completed this examination, it is now possible to make some observations on the state of the emergency organisation, confronted by the nuclear and radiological risks in France, and also on some proposals put forward for improving the arrangements made to cover the coming years.

1. The organisation, which is a pragmatic one relying on operators who know each other, designed to handle any radiological incidents which occur on the premises of users of radioactive materials and radiation sources, has reached maturity. It is equally true the organisation which covers the safety of large nuclear installations such as power for stations. The procedures have been progressively improved, and the personnel is been trained. At Electricité de France in particular, the organisation is regularly put to the test, and is found to be satisfactory.

This does not mean that the effort should be slackened. On the contrary, vigilance must continue, as must the progressive improvements achieved as a consequence of the acquisition of additional information, and of the lessons learnt by experience.

2. But there still remains a considerable effort to be made in the area of protection of people and the environment in cases of contamination beyond the limits of the large nuclear sites. Contamination of the environment, or simply the threat of it, is liable to engender serious economic and social disruption. So the socio-economic leaders should be associated with the operators, the public authorities and the experts in the nuclear field. This refers to the professions and the public industries who are responsible for daily operation of the major functions of a modern society - feeding the population, distribution of power, water, circulation of people and goods, administration of health care, and so on. The presence at the Préfecture of experts from the nuclear field and of specialists in emergency control as it is the case at the present time, is very necessary but not really sufficient. The situation will have to be dealt with by a large number of specialists from outside of the nuclear industry, and these must be prepared for the change from normal life to the exceptional state which contamination of the environment would represent.

These proposals do not represent any departure from the concept adopted at a very early stage by the representatives for nuclear safety, normally referred to as **"defence in depth"**. What is involved in fact is the extension well beyond the nuclear site of a concept which was initially created for the nuclear installation. To the successive lines of defence formed by procedures for the prevention of incidents and then accidents, and for limitation of their radiological consequences, are now added the monitoring of agricultural land and products, exemptions, adjustments concerning distribution and other reconomical measures.

The last line of defense, however, is behaviour of each individual citizen who is personally affected or who is simply listening to the news bulletins. There is no doubt that this behaviour will reflect the ability of each profession to exercise responsibility and face up to the situation.

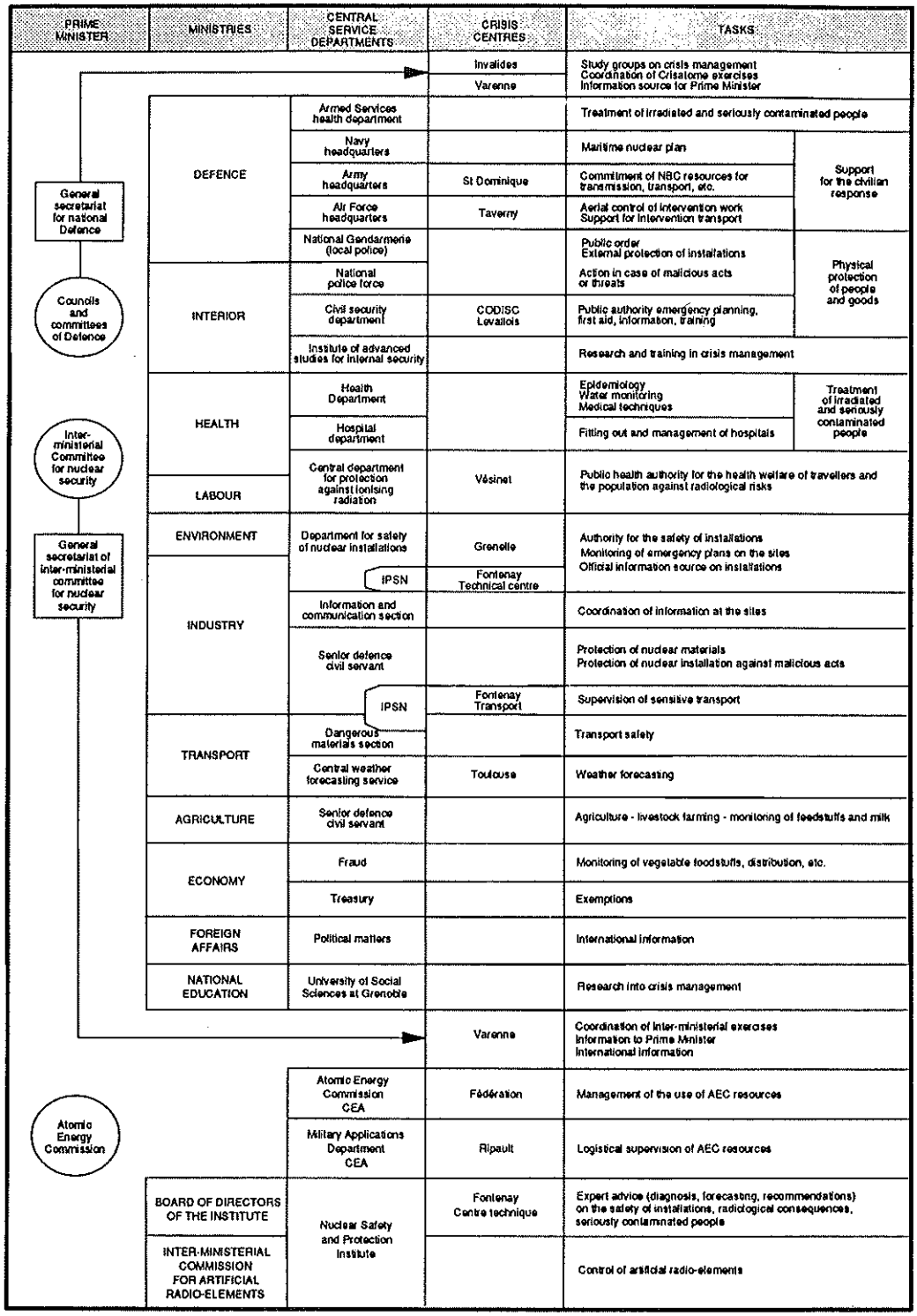

Figure 1 : SIMPLIFIED TABLE OF THE TASKS OF THE CENTRAL FRENCH PUBLIC AUTHORITIES CONCERNED WITH RADIOLOGICAL AND NUCLEAR EMERGENCIES

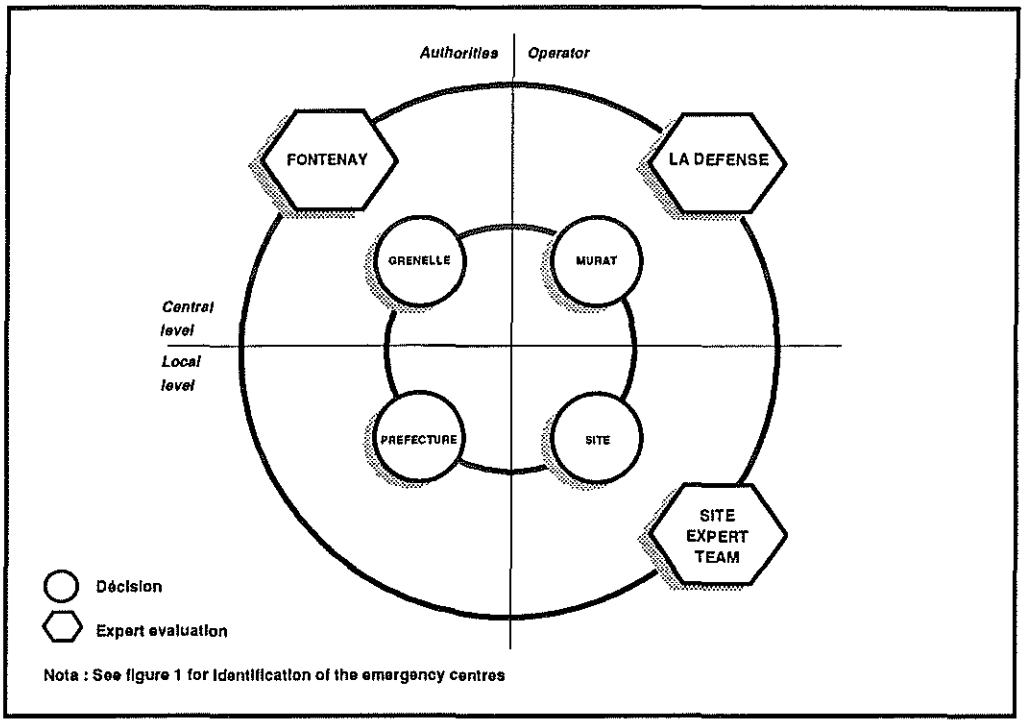

Figure 2 : RELATIONSHIP BETWEEN THE EMERGENCY CENTRES DURING THE INSTALLATION PHASE OF A NUCLEAR ACCIDENT
- CASE OF ELECTRICITE DE FRANCE POWER PLANTS -

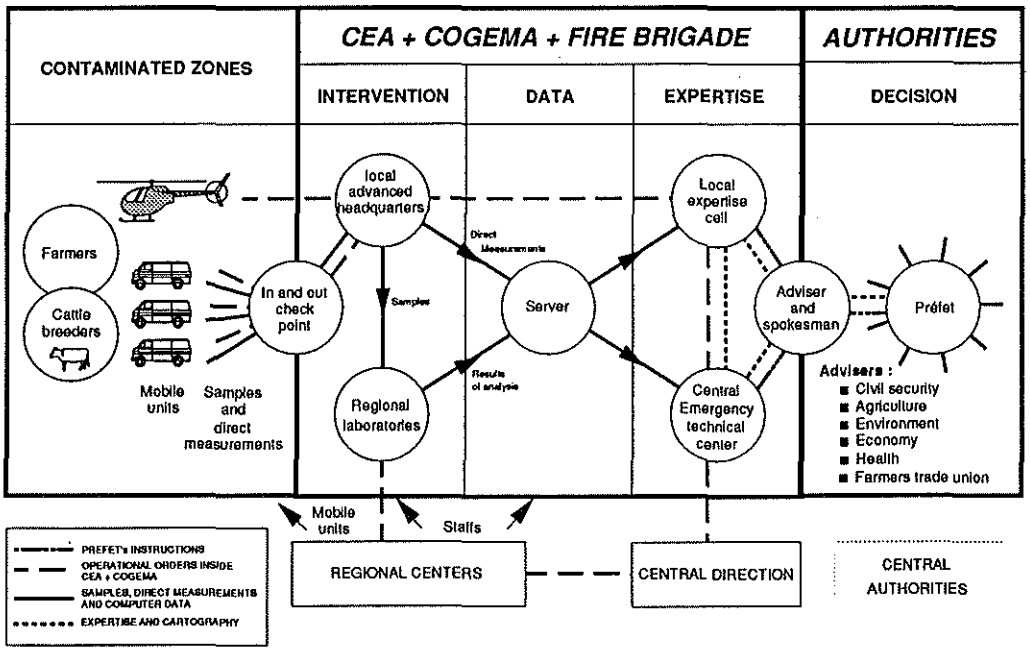

Figure 3 : FIELD EXERCISE - CADARACHE - 14-17 OCTOBER 1991

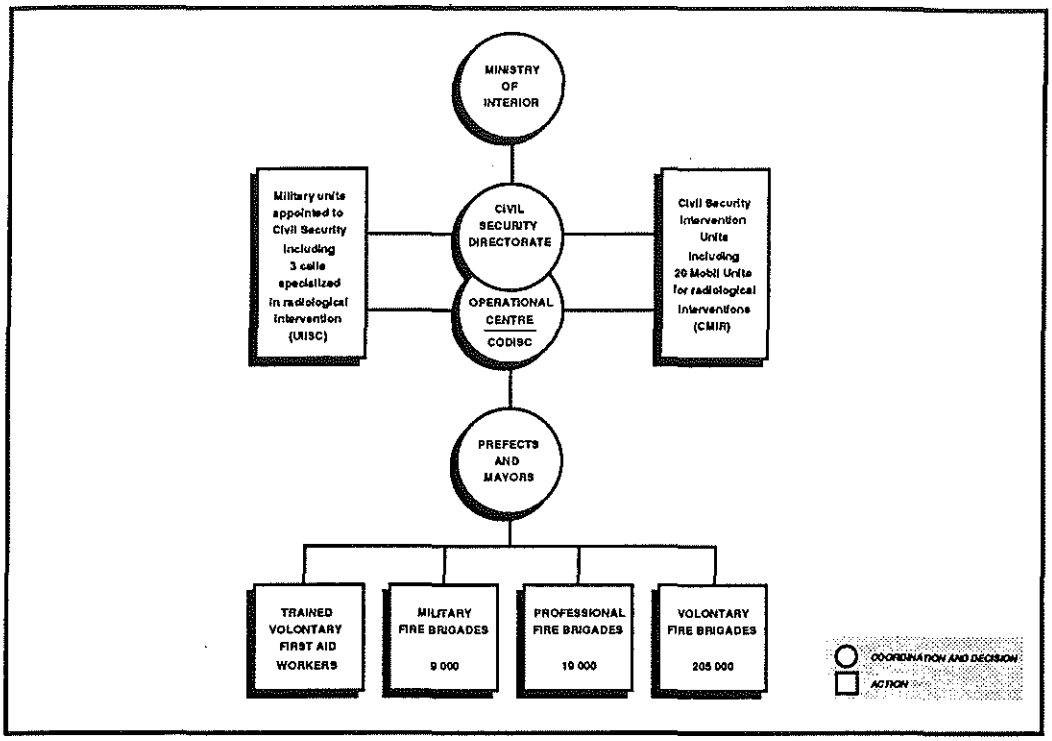

Figure 4 : ORGANIZATION OF THE INTERVENTION OF THE CIVIL SECURITY SERVICES

3rd LEVEL NATIONAL MEANS	MOBIL UNITS AND LABORATORIES FROM : ● CENTRAL SERVICE FOR PROTECTION AGAINST IONISING RADIATIONS - MINISTRIES OF HEALTH AND WORK ● ATOMIC ENERGY COMMISSION (CEA) ● COGEMA ● GIE INTRA (EDF, CEA, COGEMA) ● INSTRUCTION AND INTERVENTION UNITS OF THE CIVIL SECURITY - MINISTRY OF INTERIOR ● NBC AND OTHER UNITS OF MINISTRY OF DEFENCE ● REQUISITIONED INDUSTRIAL MEANS...
2nd LEVEL REGIONAL MEANS	MOBIL UNITS FOR RADIOLOGICAL INTERVENTION (CMIR) MINISTRY OF INTERIOR
1st LEVEL LOCAL MEANS	LOCAL LIGHT TEAMS FOR THE DETECTION OF THE RADIOACTIVITY AND MARKING OUT FROM FIRE BRIGADES, CEA CENTERS...

Figure 5 : INCREASING THE CAPACITY OF INTERVENTION

PROTECTION OF THE POPULATION IN SWITZERLAND: THE *EMERGENCY ORGANISATION RADIOACTIVITY* (EOR) AND THE *NATIONAL EMERGENCY OPERATIONS CENTER* (NAZ)

Hans H. BRUNNER
Secretary Fachverband für Strahlenschutz (FS)
c/o GSEDI/Nationale Alarmzentrale (NAZ), Postfach, CH-8044 Zurich, Switzerland

Summary

As a consequence of the nuclear weapons tests Switzerland has since 30 years expert commissions, concepts, monitoring networks, monitoring and emergency teams for the protection of the population following radiological accidents of all types inside or outside the country. Thus Chernobyl hit a prepared country - except information. The Radiological Emergency Organisation (EOR) and its National Emergency Operations Centre (NAZ) have up-to-date legal bases, concepts and operational means. Besides radiological events, NAZ deals also with chemical accidents, satellite crashes and dam breaks. Unique is the coordinated use of the combined means of civil authorities, civil defense and army in all strategic cases.

1. Development, General Concepts and Tasks

Nuclear tests caused Switzerland in 1964 to establish an emergency organisation for protection against any radiological incident inside or outside the country. The first exercise in 1968 was followed by the first active action in 1969 when a core melt occurred at the experimental nuclear power plant at Lucens not far from Fribourg - without environmental contamination and with relatively little concern to media and public...The National Emergency Operations Centre, NAZ, started as a one-man team at the Swiss Meteorological Institute with voluntary collaborators.

Some basic facts about Switzerland show the background: The political structure is similar to that of the USA: 26 cantons with their own parliaments, governments and competences in fields such as health, police, food control, civil defense and others. 5 Nuclear Power Plants occupy four sites in relatively densely populated regions. Civil defense shelters or massive basements in over 90 percent of the houses provide good shielding.

The official concepts of **Total Defense** and of **Coordinated Radiological and Chemical (AC) Protection** provide for full cooperation of the resources of federal, cantonal and communal authorities and of emergency services, army and civil defense for preparations and actions. Cantons and communities dispose of most of the means for countermeasures and actions such as health services, food control, police, fire-brigades, civil defense or public transports and must assist EOR at the order of the Federal Government.

Since 1979 the permanently operating **contact point (ARMA)** and the **National Emergency Operations Centre (NAZ**, Nationale Alarmzentrale) are located in Zurich close to the Swiss Institute of Meteorology (SMA). In 1984 a fully protected underground installation was taken into use, the permanent team of the NAZ was enlarged (now 18 persons) and supplemented by a militarized staff (Astt 800).

The *main purpose* is to protect the population, based on monitoring and intelligence, by early warning of authorities, by timely alert and instruction of the population and by initiation of protective and preventive actions.

Legal bases for the organisation are the new radiation protection law, the revised radiation protection regulations (both fully adapted to ICRP-60) and regulations on the EOR, on the NAZ and on emergency planning around nuclear installations. EOR has the legal competence to request the cooperation of any public or private specialists or services needed in an emergency.

The pre-Chernobyl action concept based on emergency reference levels has been replaced by a thoroughly revised **dose-action concept** (Table 1) which contains the general action criteria and upper and lower dose levels as action / non-action criteria both for immediate preventive actions and later countermeasures. This concept corresponding to international recommendations will be specified in detail for various types of events. Thanks to good shielding provided by massive houses and basements and to the large number of shelters in Switzerland, the main preventive action is not horizontal but vertical evacuation, i.e. stay in the house -> basement -> shelter. Horizontal evacuation would only be a later, well planned action, if necessary.

Nationale Alarmzentrale Centrale nationale d'alarme Centrale nazionale d'allarme	DOSE–ACTION CONCEPT		
Protective Action	*Dose*	*lower Dose Level*	*upper Dose Level*
Stay Indoors	eff, ext+inh	1 mSv	10 mSv
Basement / Shelter	eff, ext+inh	10 mSv	100 mSv
Evacuation, if sheltering insufficient or no longer possible / acceptable	eff, ext+inh	100 mSv	500 mSv
Other Protective Actions	eff, ext+inh	1 mSv	500 mSv
Intake of Stable Iodine	Thyr, Inh, Iod	30 mSv	300 mSv
Restrictions Food	eff, Ing	1 mSv	20 mSv

Effective Dose first year after event, without action considered but effect of previous actions included

Table 1: Dose-Action Concept

A fundamental concept is to *use normal means and structures as far as possible, and extraordinary ones only where necessary*. Regrouping in the early phase of an event could only result in chaos. Governments, authorities, public services on all levels must have an emergency organisation allowing them at any time to cope with unusual events besides their normal operations. Special means needed for emergency purposes, such as computers and monitoring networks, are regularly used for normal purposes, too.

2. The Emergency Organisation Radioactivity (EOR)

The Radiological Emergency Organization (now EOR) operated very successfully after the Chernobyl accident but was then restructured to improve the cooperation at the government and political levels and the information of the public (Fig. 1). Until Chernobyl the Federal Commission for Radiological and Chemical Protection (now KOMAC) was in charge of the emergency organisation, and its chairman was also the commander of EOR in action. After Chernobyl the EOR was directly subordinated to Government and the **General Secretary of the Department of the Interior** (GSEDI) is now the chairman and the commander of EOR. He is supported by a **Managing Committee** (Leitender Ausschuss Radioaktivität **LAR**) consisting of the directors of the relevant federal agencies. This links the Emergency Organisation directly to the Government. The **Information Centre** of the Federal Chancellory is responsible for communications with the cantonal authorities and for information of the public.

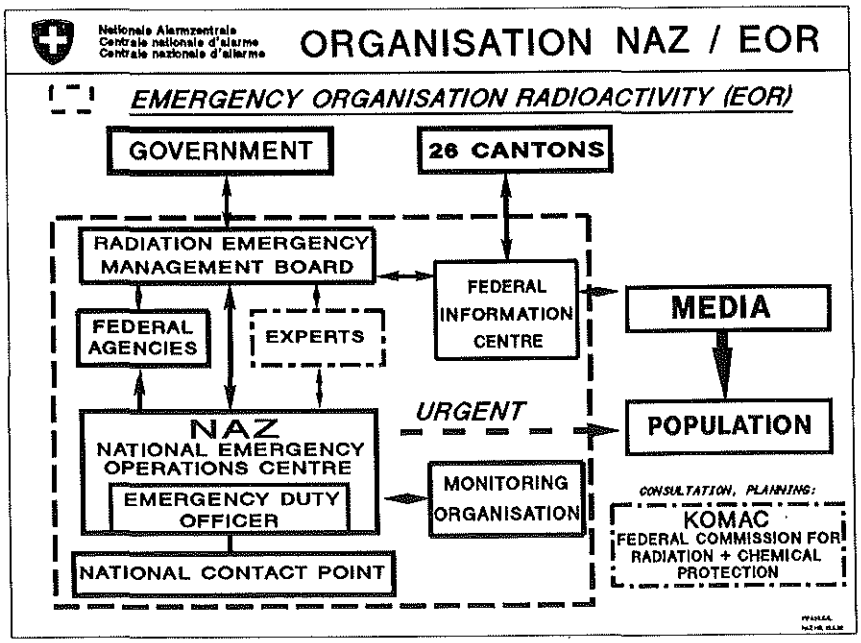

Fig. 1: Structure of the Emergency Organisation Radioactivity (EOR)

The **Federal Commission for Radiological and Chemical Protection (KOMAC)** with 25 members and an additional 70 scientists in its Working Groups has taken a purely advisory but nevertheless indispensable role (Fig. 2). It supports the NAZ and EOR in the development of concepts, in the preparations and in the conduct of exercises.

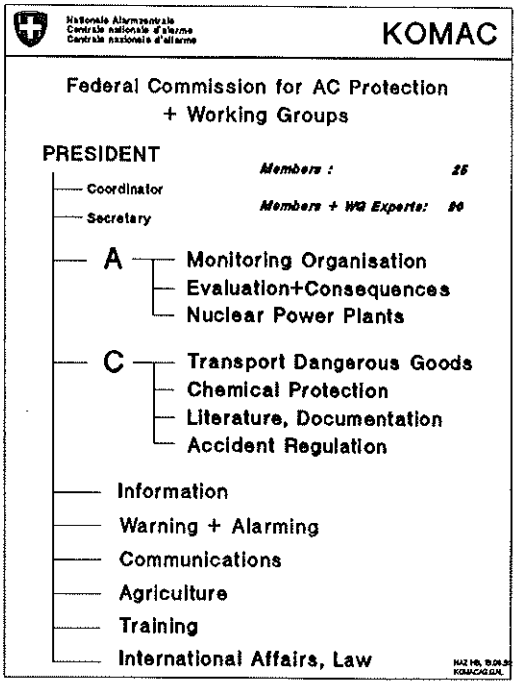

Fig. 2:
Federal Commmission for Radiological and Chemical Protection (KOMAC) and its Working Groups

3. National Emergency Operations Centre (NAZ)

The NAZ forms EOR's permanent nucleus and first-stage action team. A **contact point** operating 24 hrs/day (**ARMA**) at the meteorological computer center, a **duty officer** and **automatic monitoring networks** assure instant readiness, rapid warning and alarming of authorities, emergency teams and population, followed by evaluation of monitoring data and quick initiation or proposals of protective and preventive actions.

The normal tasks of NAZ, in cooperation with KOMAC, federal and cantonal agencies, are the preparation, coordination, maintenance, improvement, training and testing of the EOR and of its elements.

NAZ deals with a large *range of events*: radiological events, from transport or operational incidents over nuclear installation accidents to nuclear weapons events in peacetime and war, but also Nuclear and conventional satellite crashes, chemical incidents (only advisory role) and dam breaks.

4. NAZ and EOR in Action

In action, the NAZ operates from an autonomous protected installation with modern data processing and communications systems and safe communications to national partners, neighbouring countries and IAEA.

The operational structure in action is centred around the three overlapping sectors **intelligence + information / monitoring / evaluation and actions**, which are supported by technical, computer and communications services. (Fig. 3)

Within a few hours the permanent team of the NAZ can be enlarged by a **military staff** (Armeestabstell) of over a hundred scientists (mostly members of KOMAC working groups), specialists, and support personnel. Thanks to the militia system of the Swiss Army, they have been recruited among the best specialists and scientists in radiation protection, nuclear safety, chemistry, data processing, communications and other related fields. Their main tasks are the management of the monitoring and laboratory organisations, the collection and evaluation of all informations and data on the event and its consequences, the presentation of

up-to-date informations on the situation , especially the radiological one, the evaluation and proposal of protective actions to be taken, the judgement of their effectiveness, and whatever consultative support may be required by the federal authorities and the Information Centre. Thus the militia system and compulsory reserve duty in army and civil defence allows us to make use of experienced scientists and specialists during 20 days per year for preparations, training and exercises and at any time in a real emergency. Many of these specialists are also members of the advisory Federal Commission (KOMAC) and its working groups. They are motivated to provide practicable solutions because they themselves may have to use under emergency conditions the concepts and means they helped to prepare.

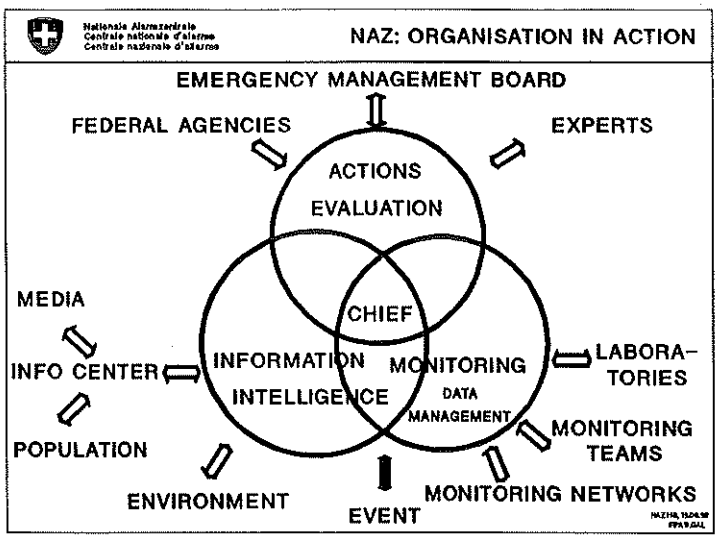

Fig. 3: Operational structure of NAZ

5. Monitoring, Data Processing, Communications

The **monitoring organisation** shall permit to detect, identify and verify radiological events and hazards, to forecast, follow and evaluate their development and extent, to estimate the doses to the population and to check the efficiency of countermeasures. It consists of automatic networks and of quickly activated monitoring posts, a laboratory organisation, and mobile monitoring and sampling teams with cars and helicopters. Details are described in several other papers at this seminar [1, 2].

The main **monitoring network** is NADAM, an automatic radiation monitoring network coupled to the automatic meteorological monitoring network. It collects dose rate and weather data every ten minutes from 58 stations all over the country. Daily dose-rate averages from 16 stations are shown on text-TV together with basic informations on radiation and nuclear safety [1]. Other automatic networks collect and measure aerosols (FWP/RADAIR) or monitor the environment of nuclear power plants (MADUK). 107 monitoring posts (AWP) at police stations, equipped with highly sensitive dose rate monitors, can be activated within 1-2 hours and transmit data at short intervals. Mobile monitoring teams comprise special teams of health physicists with all sorts of instruments and sampling devices, and teams of police, fire-brigades, army and civil defense radiation protection specialists, monitoring helicopters and an aeroradiometry helicopter.

The core of the **laboratory organisation** for sampling and analysing all kinds of samples and especially food are 5 permanent special laboratories which also do routine radiological environmental monitoring. 20 cantonal chemistry laboratories have been equipped with radioanalytical instrumentation and can on short notice be supplemented by 30 total-defence radioanalytical laboratories with militarized staff in shielded underground locations. This suffices to collect and analyse the large number of daily samples required to control food quality all over the country, making use of established food control routines. Several stationary whole body counters were already of good use after Chernobyl; additional mobile ones are planned.

If we activate the full means of army and civil defense in addition to the normal means of the EOR we would have more than one radiation monitoring instrument and trained staff on each of the 41'000 square kilometres of Swiss territory, and a laboratory for food control per 1000 square kilometres.

Chernobyl provided invaluable experience about the size and types of *data* which had to be received, stored, evaluated and presented. On this basis a **concept of communications and informatics** was worked out which is almost completed on the hardware side and quite advanced in the software preparation. All the monitoring data (from routine monitoring and in emergencies) are transmitted to and stored and analysed in a large data processing system at NAZ (project **PHOENIX**) which is also the base for several applications and expert systems, for visual presentation of results (ARGIS) and for the exchange of data and informations within the EOR and its partners. [1, 3]

Communications must be reliable and redundant in all situations and make use of all available technical means and public as well as special systems, including those of the army. Dedicated lines for phones and fax are installed to NPP and nuclear safety authorities, among others. An electronic mail system, VULPUS Telematic, has replaced the telex network linking NAZ and all cantonal police commands. A video installation allows video-conferencing between NAZ and the LAR command post in the capital. Both for data processing and communications simple robust stand-by methods must be available in case of failures of the main systems.

Monitoring is only part of the entire *intelligence activities* needed for detecting, verifying and evaluating an event, especially if it happens abroad. An extensive information network with links to news agencies, military intelligence and private information networks such as NucNet, and close contacts to partner organisations in neighbouring countries, institutions and scientists in other countries and to international organisations such as IAEA and ESA are indispensable for NAZ. IRPA and its associate societies, especially the Fachverband für Strahlenschutz, are of invaluable help in this task.. NAZ uses a radio and TV monitoring installation allowing permanent monitoring and selective recording of dozens of TV and radio programs.

6. Emergency Preparedness Around Nuclear Installations

The main nuclear installations in Switzerland, 5 power reactors and a nuclear research center (PSI), are concentrated in three regions: KKL, KKB and PSI share the lower Aare valley near the Rhine and fall into a joint set of *emergency planning zones*. The other two regions are KKG between Aarau and Olten and KKM east of Berne (Fig.4). Based on the pessimistically evaluated effects of a **Reference Accident** each region consists of two concentric zones around the sites [4]. The inner **Zone 1** with a radius of approx. 3-5 km corresponds to the area where the cloud dose from a release could cause acute health hazards to persons remaining outside, while in the outer **Zone 2** with a radius of 20 km the cloud doses might be above the lower level of the dose action concept. The rest of the country outside the zones 2 is also called **Zone 3**. The NPP operators, the site cantons and the communities in the zones 1 and 2 together with the EOR/NAZ and the Nuclear Inspectorate (HSK) are the main elements in this emergency preparedness [4].

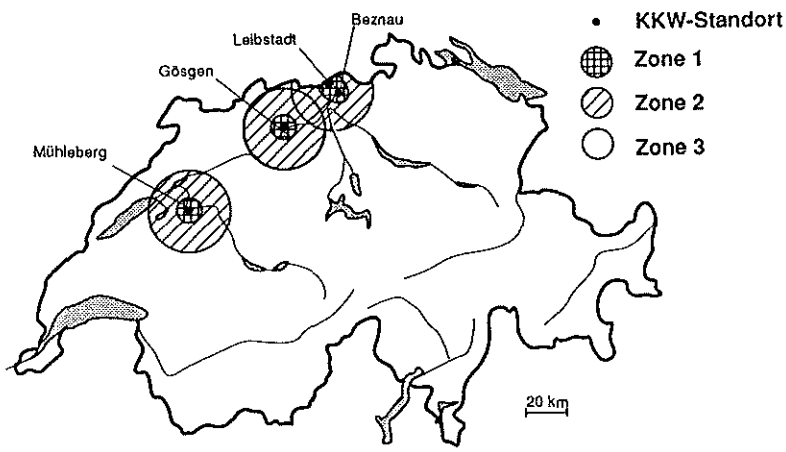

Fig. 4: Sites and Emergency Planning Zones of Swiss Nuclear Power Plants [4]

The required minimum reaction times of NAZ and EOR elements are also determined by the reference accident. **"Warning"** means alerting the authorities in the zones 1 and 2, the site cantons and the EOR early enough during a developing accident that they can get their forces ready for action before a release actually happens. "Warning" is transmitted by dedicated phone systems and telefax/telex, to zone 1 directly from the NPP, to zone 2 and the others by NAZ. When it becomes necessary to activate the population, **"General Alarm"** is sounded by means of the sirens installed more densely in zones 1 and 2 but existing in all communities in the country. The population is instructed that this means "turn on your radio", and it is a task of NAZ to provide coincident *radio messages* giving information and instructions. The sirens are tested once a year; the instructions to the public are found in every phone directory and in pocket agendas etc.

Zone 1 is always alarmed entirely. For zone 2, depending on the weather situation, NAZ can select one or several overlapping 120° sectors where the alarm takes effect. Population size in these sectors around a NPP may vary by a factor of up to 10. Before or at the beginning of a release another "General Alarm" or a special **"Radiation Alarm"** can be triggered to order the population to hide in basements or shelters. The entire warning-alarm-sheltering sequence must be completed within less than 4 hours after initiation of an accident sequence at a reactor.

After a Warning NAZ/EOR would dispatch several elements of the **monitoring organisation** to a "Monitoring Command Post" (MLS) near the boundary of zone 2 from where monitoring helicopters and mobile teams would take a predetermined set of background measurements and samples from Zone 1. When a release begins, a monitoring helicopter would follow and locate the cloud. After passage of the cloud aeroradiometry and mobile ground teams would start to asses the extent of the ground contamination and collect site-dosimeters and samples of soil, vegetation etc. for analysis in the special laboratories. The contamination map thus composed serves as a basis for further decisions about protection against external exposures. Together with the samples taken it will allow a forecast of internal exposures to be expected, thus allowing a timely start of countermeasures in the agricultural and food sectors.

Federal regulations on **iodine prophylaxis** have just been implemented this summer and stable iodine tablets will be distributed in the coming months to all households, schools, workingplaces and public buildings in the zones 1 and in most parts of the zones 2. For zone 2 the cantons have the option to distribute the tablets only to neighbourhood distribution points such as drugstores in place of to the households. The tablets for the rest of the population, i.e. zone 3, are centrally stored by the cantons. NAZ is responsible for advising the population when to use the iodine tablets.

7. Information, Public Relations

Information was the weak spot after Chernobyl, not on the scientific and specialist level but on the political and government one. As a consequence the **Information Center (Infozen)** of the Federal Chancellory was made fully responsible for this task and was accordingly reorganized and equipped. The facts will come from NAZ and EOR which will provide consultants to the Infozen. The responsibilities for information in various cases for events at Swiss or foreign NPP have been settled.

But information of the public in an emergency is only credible and accepted, if NAZ and EOR are already familiar to the public and to local and regional authorities. This requires a well planned continuous *public relations program* which makes use of all publicity means and of all situations where the media and the public may be willing or interested to learn about emergency preparations. The Gulf war in 1991 and the St.Petersburg NPP incident in spring 1992 were such occasions. The wide-spread radiophobia among media, politicians and public and their tendency to rather believe horror stories than facts do not simplify this difficult task.- The excessive fear of radiation and the tendency of the media to rather accept exaggerations and sensations than facts, are additional complications and require a lot of information work even for far away nuclear incidents.

In a multilingual country with three official *languages*, with 15% alien population and many tourists using several dozen languages the information is further complicated by the need for translations.

8. International Cooperation

As a small country surrounded by nuclear countries and with our own NPP near the borders we have **bilateral cooperation and information agreements** with Germany, France and Italy. We have close cooperation with IAEA and other international organisations such as NEA, CEC and ESOC (Fig. 5). Thanks to IRPA we keep excellent contacts to most other European and some overseas emergency organisations, radiation protection specialists and institutions.

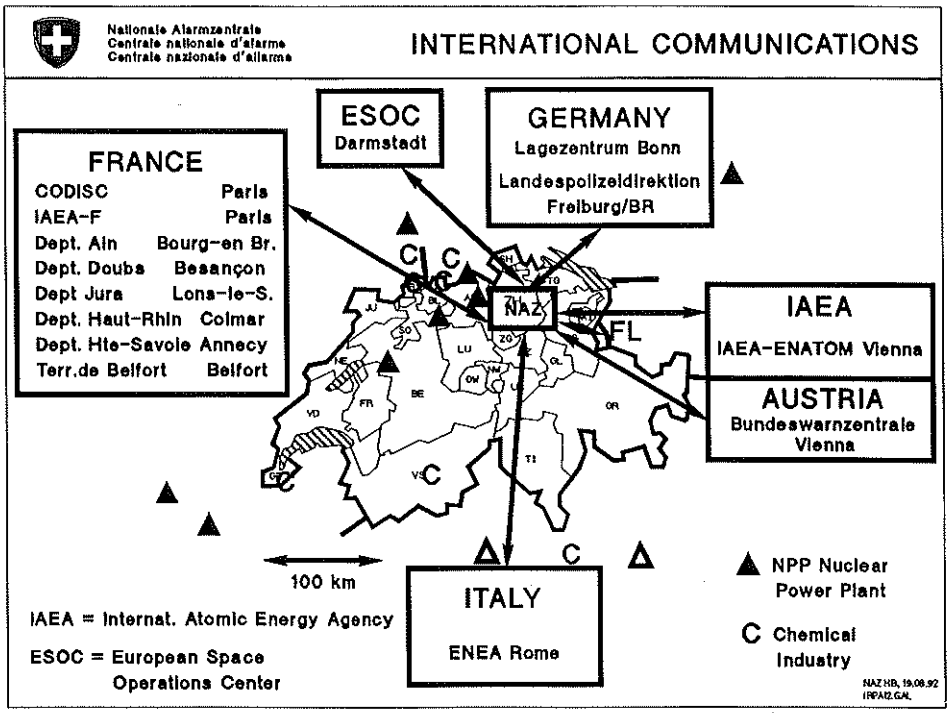

Fig. 5: International Communications of NAZ

In a situation such as the recent NPP incident at St. Petersburg, Russia, where everybody was asking for information, we could establish excellent contacts to our Scandinavian colleagues and quickly receive verified informations and data.

9. Training and Exercises

A *training concept* is in preparation for a coordinated and systematic basic and advanced information and training of the various elements and levels of the EOR. Close cooperation is established and will be improved with all existing training institutions in the nuclear, radiation protection and emergency preparedness fields.

Exercises of parts or the whole organisation are done every year. They range from table-top or staff exercises to full scale field exercises involving most elements of total defense. NPP are required to have an emergency exercise each year, every third year involving the EOR and the cantonal and communal staffs. The next such exercise will be in February 93 at Mühleberg. Their preparation, execution and evaluation put a large additional load on the NAZ, so certain standardized exercises are in preparation which can be easily used and adapted for exercises with parts of the EOR or with regional or cantonal staffs.

10. Experiences and Problems

As perfect as this organisation may look, we have not yet solved all problems, and new ones are waiting.

Chernobyl was not the first but the largest real operation of the EOR. Although it was no genuine emergency as far as exposure levels and doses were concerned, it allowed a full-scale run of the monitoring and laboratory organisation and data evaluation far beyond anything feasible in exercises. Political decision making and information were put to a severe test and became the main areas for improvements.

The permanent team of NAZ, expecially the contact point ARMA and the duty officers have to deal with several small operational and transport accidents every year and distribute information on radiological events in other countries several times per month. these minor events are good tests for start-up procedures, communications and checklists.

It is not easy for NAZ to find qualified permanent collaborators with good professional training, experience, flexibility, fluent in several languages and willing to serve as duty officers several weeks per year.

As everywhere in the nuclear and radiation protection field we also have an abnormal age-structure in ou staffs and will have to replace more than half within the next 5-10 years. How to find sufficient new and we trained staff when radiophobia makes any nuclear or radiation job unattractive, is a yet unsolved problem as the transfer of all the knowledge and experience accumulated by the first generation to the newcomers. Ho far artificial intelligence will be able to ease that problem remains to be seen; we are working on prototypes fo **decision support systems** (DSS) [3].

Although Chernobyl set free an unexpected amount of funds for emergency preparations, this effect may soo decay, so all new investments will need a thorough justification and optimisation.

European integration and cooperation will increase in the near future, and our concepts as well as monitorin data processing and communications systems must be or become internationally compatible.

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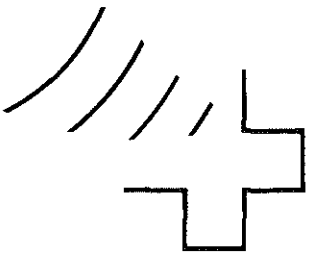

Nationale Alarmzentrale
Centrale nationale d'alarme
Centrale nazionale d'allarme
Centrala naziunala d'alarm

CH-8044 Zürich, Postfach

EMERGENCY PROTECTION AS CONTROL LOOP

A. Bayer
Bundesamt für Strahlenschutz
Institut für Strahlenhygiene
Postfach 1108
D-8042 Neuherberg-München

Summary:

Emergency protection is a control loop that includes the following elements: disturbance device "release of radionuclides and contamination process", control process "monitoring of contamination and radiation exposure by measurement and model calculation", control program "recommendation of limits; evaluation and decision process", and controller "taking of countermeasures". The essential components of these elements are described. For a simulation of this cybernetic system it is necessary to transform it into a model or system of connected models. This system is then used for planning and exercising and in the event of an emergency for finding an optimum countermeasure strategy.

1) Introduction

The objective of emergency protection is to limit, as far as achievable, all possible consequences of an emergency, i.e. an accidental release of radionuclides into the environment, by decisions and actions of technically qualified and competent personnel and by use of installed equipment and stored devices and materials. Emergency protection is therefore a planning science with the general aim to find ways for the systematically and optimally apply all available means - also those still in the planning stages - in accordance whith the concept designed to achieve a defined result.

Essential components for controlling an emergency are the monitoring of environmental radioactivity and the taking of measures for the prevention and limitation of possible consequences. These are the elements which constitute a control loop together with the accidental event including the subsequent contamination process and the protection plan (legal provisions, etc.) as well as the "evaluation and decision process". This loop represents a cybernetic system with elements that are functionally interrelated and react to external influences. The above defined control loop is shown in Fig. 1.

2) Emergency protection as control loop

The elements can be assigned to the control loop "emergency protection" as follows:

- The "disturbance device" represents the contamination process after an accidental release of radionuclides into the environment; this is the ≪interference process≫.

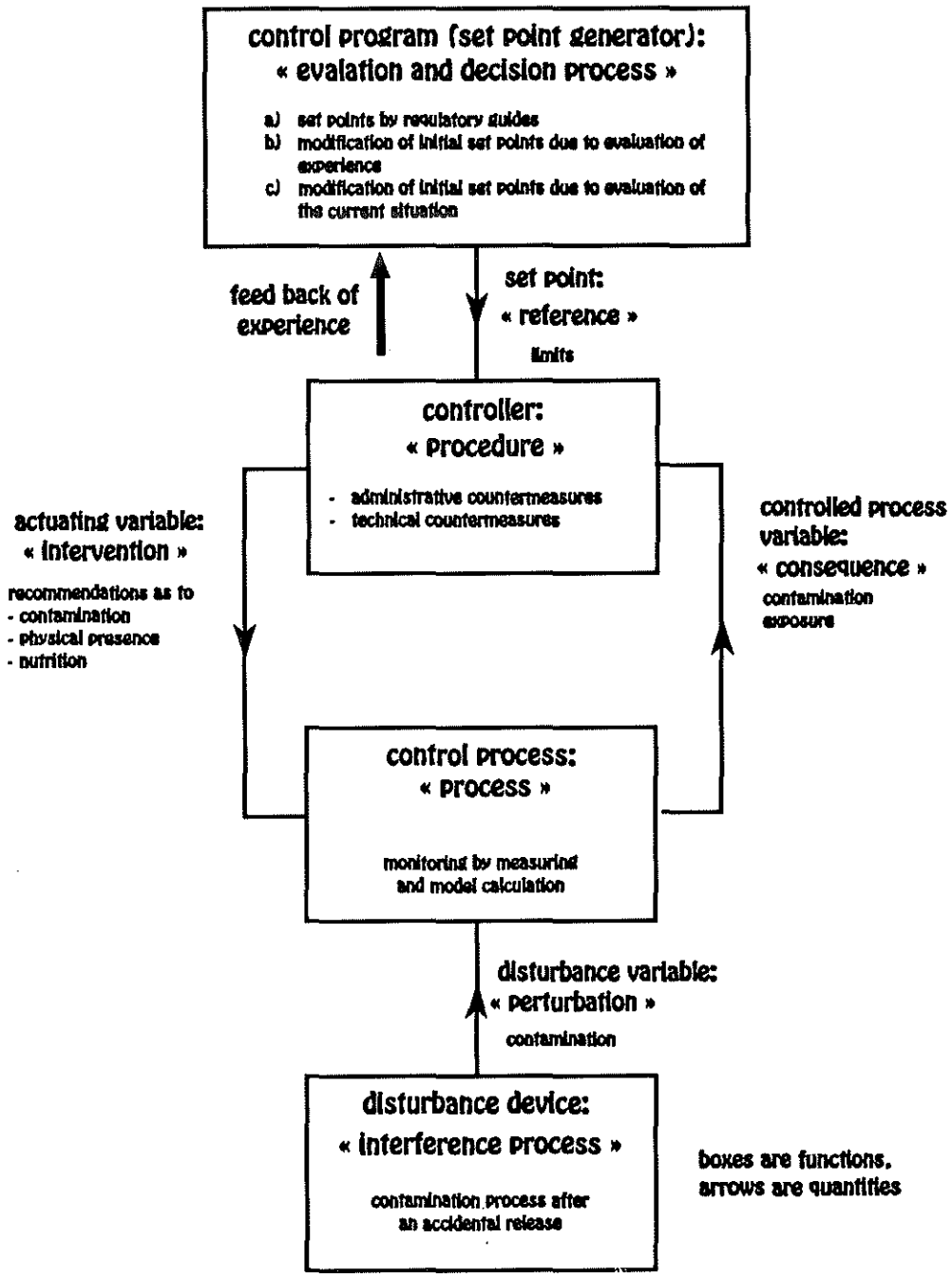

Fig. 1: Control loop "emergency protection" (adaptive control loop)

- o The "disturbance variable" represents the resulting contamination (≪perturbations≫).

- ■ The "control process" is the monitoring process, which includes the monitoring of contamination and radiation exposure by measurement and calculation on a model. The actual condition is determined and the future condition predicted; this is the ≪process≫.

- o The "controlled process variables" contamination and exposure are determined by monitoring (≪consequences≫).

- ■ In the "control program", set points are prepared on the basis of regulatory guides (described as function a in the control program); this is the ≪evaluation and decision process≫.
  An evaluation of the experience (gained either at home or abroad) suggest the practicability of a continous modification of the "control program" (described as function b in the control program). During or subsequent to a contamination, it may also be useful to modify the control program after an evaluation of the current situation (described as function c in the control program). These modifications of the "control program" convert - in cybernetic terminology - the control loop into an adaptive control loop.

- o The thus established "set points" are then entered into the control loop (≪references≫).

- ■ On the basis of the estabished controlled process variables, the "controller" recommends countermeasures; this is the ≪procedure≫.

- o The countermeasures recommended by the controller concern changes in contamination, physical presence, nutrition etc. These are the "actuating variables" (≪intervention≫).

The elements of the control loop must be interlinked by an information system that guarantees the immediate transmission of any information between and within the elements. This is the system that enables the functioning of the control loop.

3) Simulation of the control loop "emergency protection"

The pattern of the control loop or of its elements for purpose of simulation leads to arithmetic-logical, dynamic models. These are used for imitating the chronological development of all relevant system state variables.

The models developed and used in the control loop "emergency protection" are serving different purposes. On the one hand models are used that are capable of additionally fullfilling tasks in an emergency which essentially cannot or cannot yet be done by

measuring or other facilities at certain times. Examples of the first situation are models for determining the radiation exposure on the basis of the contamination determined in various environmental regions. Examples of the second situation are models for the prediction of contamination from an approaching cloud.

On the other hand models are used that simulate all processes that occur during an event, as, e.g., emission, contamination, information processing and transmission, decisions, countermeasures. The models from the first group naturally apply here as well.

The situation is demonstrated in Fig. 2.

In general, models and simulation are used for

- an arithmetic-logical description of the system, also of those components of emergency protection, which in the majority are often described only verbally.

- examining the optimum planning of facilities as well as stored devices and materials.

- examining possible accident scenarios and subsequent countermeasures, in particular in view of necessary improvements in the control loop but also in view of improvements in facilities with radioactive inventory.

- examining the integrity, the unequivocal or uncontradictory quality and efficiency of the decision process (responsibilities, flow of information).

- investigating the influence of insufficient or lacking knowledge, especially in view of the extent in which the level of knowledge needs to be improved to effectively reduce the uncertainty of a consequential decision.

- examining in case of an event the consequences and the practicality of a decision

Moreover, a complete model of a control loop "emergency protection" is an excellent exercise for all individuals working in the control loop and thereby experience the reaction of the system to different decisions.

In case of an emergency these tools serve to find an optimum countermeasure strategy by using the entire system to reach the intended goal of "least possible consequences".

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Acknowledgement:
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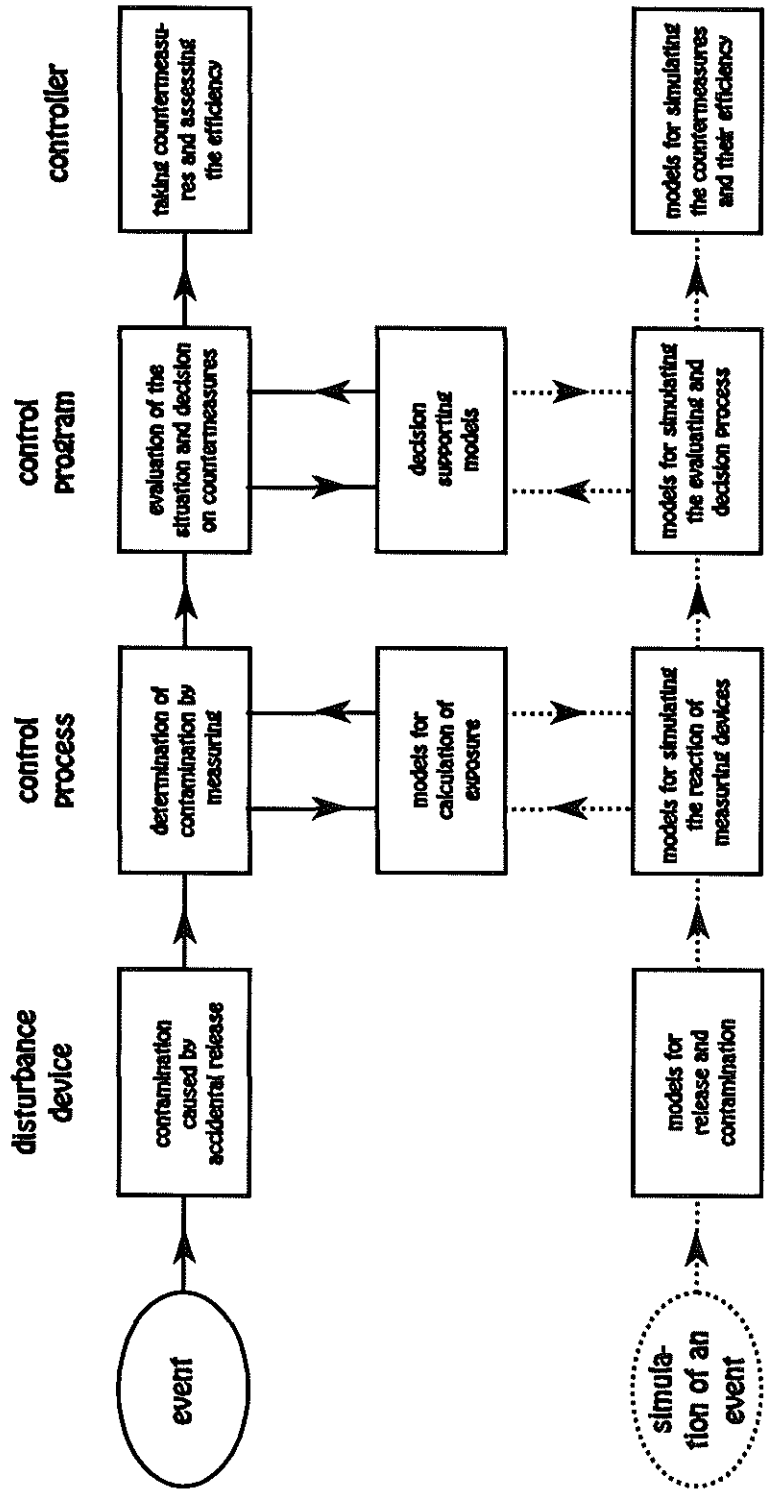

Fig. 2: Use of models in the case of an event or simulation of an event (simplified)

METHODS AND TOOLS FOR THE RAPID EVALUATION OF THE RADIOLOGICAL CONSEQUENCES OF AN ACCIDENT

Mme Sylvie Rénier,
M. Philippe Hartmann, M. Jean-Bernard Tessereau

Electricité de France - 93203 Saint Denis, France

Abstract

The difficulties encountered during emergency exercises, intended to evaluate releases of radioactivity into the environment, have led Electricité de France to develop, in collaboration with the Institute for Nuclear Safety and Protection, a **method** and **simple tools** intended specifically for local emergency teams.

The method on which the tools are based has a three-step structure, and involves:

- determining from the outset of the accident, the upper bound of foreseeable releases,

- following the development of the releases using the measuring circuit fitted in the stack,

- interpreting the readings taken in the environment.

The tools consist of a "paper" document and a computer software package.

1. INTRODUCTION

When, in a nuclear power plant, an accident arises which could bring about releases of radioactivity into the environment, the operator should be able to provide the government, in as short a time as possible, with information concerning the radiological consequences of foreseeable or actual releases.

This information is essential in allowing the government to decide on any protective measures to be taken with regard to neighbouring populations.

The exercises carried out within the framework of the crisis organisation have shown that the persons responsible for evaluating releases into the environment (at the source) did not possess sufficient means for making the forecasts. These forecasts were in fact established from very conservative results taken from accident studies indicated in the nuclear power plant safety analysis reports. Furthermore, calculations

of the radiological consequences (dose equivalent received by the public), were carried out manually using field charts, which required a great deal of time and were a source of error.

These difficulties have therefore led Electricité de France to develop, with the collaboration of the Institute for Nuclear Safety and Protection, a method and simple tools which allow the emergency teams to perform evaluations of gaseous releases and their radiological consequences on the environment with greater ease and rapidity.

2. EMERGENCY METHOD AND TOOLS

The evaluation method and the tools are based on the principles laid down by the emergency health plan published by the Curie Institute and the recommendations of ICRP publication No. 40 concerning the protection of populations.

2.1. The Method

The evaluation of the radiological consequences of an accident is carried out according to a method with a three-step structure. These three steps consist in:

1. Determining, from the first phase of the accident, the upper bound of the foreseeable releases. Given the importance of the repercussions that the implementation of certain protective measures can entail, it is of primordial importance to possess as rapidly as possible the **orders of magnitude** of the potential releases and doses which could be delivered during the accident. Such evaluations can only be obtained within a short length of time if the emergency teams possess predetermined model data for a sufficient number of accident situations. These predicted values are compared with levels specified in ICRP 40 concerning the protection of populations (sheltering and evacuation).

2. Monitoring the development of releases escaping via the stack, and then, from this, calculating the radiological consequences as a function of the meteorological conditions at that moment.

   These results are then compared with the aforementioned predicted values.

3. Interpreting the first results of readings taken in the field.

   These readings taken in the vicinity of the site are compared with the calculated results. In this way the predicted evaluations can be confirmed or adjusted, and/or an abnormal leak from the containment which has not been collected by the ventilation circuits can be detected.

2.2. The Tools

The tools consist of:

- a "paper" document detailing, specifically, those elements necessary for the determination of the quantities of radioactivity released (source term),

- software operating on a PC-compatible microcomputer intended for use in calculating the radiological consequences of releases.

The "paper" document includes several headings which make it possible to apply the method described above.

1. Predictive evaluation

The potential releases are calculated *a priori* for nine accident situations. The calculations are based on hypotheses which take account of the state of the fuel, the state of the containment and the availability of certain items of equipment such as the containment spray system and the iodine traps in the ventilation circuits. These situations are diagnosed during the accident using the logic diagram given in Figure 1, see below.

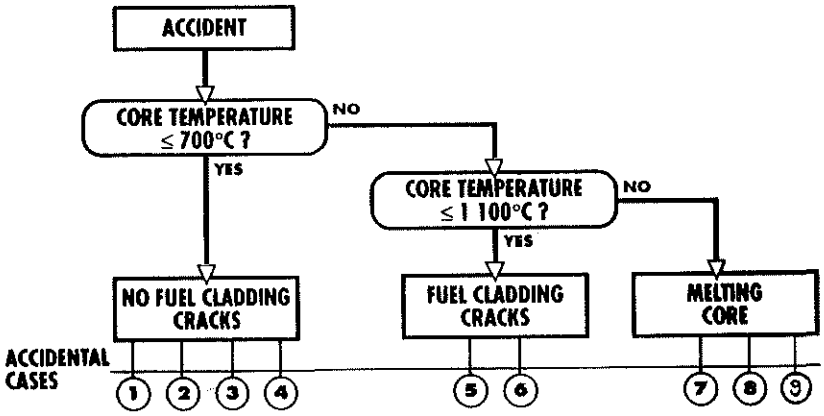

Figure 1

For each of these states, the radiological consequences of the releases are calculated *a priori* for the three most representative meteorological conditions.

The **orders of magnitude** of the potential releases and of the radiological consequences of the releases are entered in forms, and can therefore be obtained without the need for calculation.

However, if a major event modifies the course of the accident, and therefore the calculation hypotheses (e.g. loss of coolant possibly leading to cladding bursts or to melting, implementation during outage of the spray system, loss of containment etc.) the initial prediction should be readopted.

2. Monitoring the releases escaping via the stack

The document provides an outline of the calculation making it possible to calculate the amount of radioactivity released via the stack by integrating variations in the stack instrumentation signal.

The radioactivity of iodine and other airborne particles is determined either by readings taken by the sampling devices (carbon cartridge and filter paper), or by calculation, using coefficients which alter depending on whether the spray system and the iodine traps have been put into operation.

The radiological consequences of the releases determined in this way are calculated using a software program* which makes its calculations based on the distance (up to 10 km) and on the local meteorological conditions:

- the whole body dose caused by the passage of the radioactive cloud,

- the dose received by the thyroid gland through inhalation of iodine,

- the dose rates caused by deposits.

3. Interpretation of readings taken in the field

The external exposure caused by the plume can be determined by the dose rate readings (ambient gamma radiation) taken by the site monitor or downwind using portable apparatus.

Internal exposure through inhalation is evaluated by airborne dust radioactivity readings (filter papers and active carbon). Each station has two vehicles fitted with the necessary equipment.

The "paper" document provides the **orders of magnitude** of radioactivities which can be measured in the vicinity of the site, depending on the nature of the accident.

Outlines are provided which also make it possible to make

*This software program will be replaced before 1993 by a computerized system featuring a more precise diffusion model called a "puff model".

manual calculations of the ambient radiation (μGy/h) and of the atmospheric contamination (Bq/m^3) simply and rapidly, from the stack releases, and therefore to compare the results obtained from the calculation with the values measured in the environment.

The readings and the evaluations made during the accident are subsequently noted on diagrams (Figure 2).

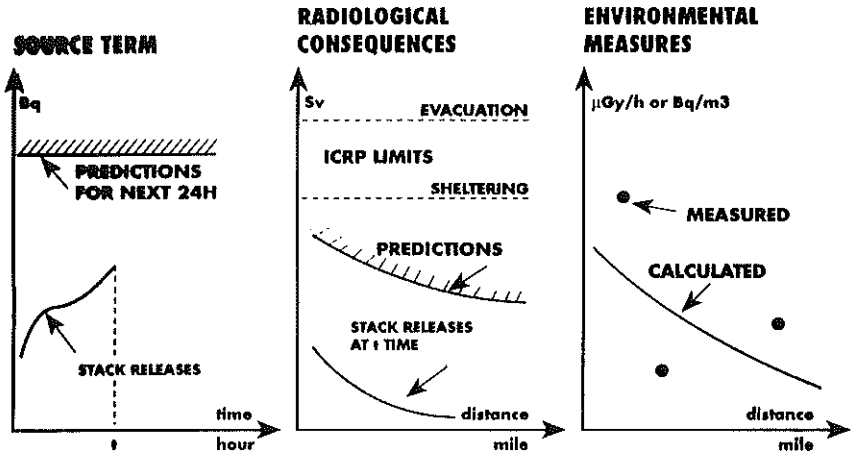

Figure 2

CONCLUSION

This method and these tools have been tested in emergency exercises. Their efficiency has been proven in the area of making early evaluations (up to 24 hours after the accident).

These tools, which were specifically designed for nuclear power plant technicians, are in no way intended to replace the work of experts, particularly of those who would be present in the national emergency centres and who would have at their disposal more precise computer tools for determining the real impact of the accident.

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DECISION-ORIENTED MAPPING IN EMERGENCY AND POST-ACCIDENT SITUATIONS

M. KANEVSKY (1), V. KISELEV (1), P. FACHE (2), J. TOUCHE (2)

(1) Institute of Nuclear Safety (IBRAE), B.Tul'skaya 52, 113191 Moscow Russia
(2) Institut de Protection et de Surete Nucleaire, Departement de Protection de l'Environnment et des Installations IPSN/CEA BP6-92265 Fontenay-aux-Roses Cedex France

Summary

The concept of decision-oriented mapping (DOM) in emergency and post-accident situations in nuclear engineering is presented. The DOM is an integrated result of decision support systems (DSS) and geographic information systems (GIS). Short survey of systems used for emergency and post-accident situations by IBRAE and IPSN are presented.

1. Introduction

The assesment of the impact of a nuclear accident on the environment, in the purpose of decision making, needs:

-to order a big amount of data on various topics like demography, geology, hydrogeology, agriculture, into graphical or alphanumerical databases and to use them with various mathematical models, concerning the transport of radioactivity in atmosphere, soil, water systems etc: this is the goal of ENVIRONMENTAL DECISION SUPPORT SYSTEM (EDSS) [1].

-to synthesize and prepare results immediately acceptible by non-specialists. The best way to do it is to realize special maps, more oriented towards decision than standard maps, overlayed with another layers of information, devoted to logistic actions: these operations can be easily made with GEOGRAPHICAL INFORMATION SYSTEMS (GIS).

The answer to the complexity of these problems is the use of computer power. The first reason is obviously for running models. The second reason is the recent development of infography that allows linking pictures to results of models, storage of large amount of data, speed of data processing, flexibility of output (variable scales, choice of accuracy, level of details) suited with each user's needs.

The present report is devoted to the complementary concept of EDSS and GIS, for their use in DECISION ORIENTED MAPPING (DOM) on nuclear installations, in case of Emergency and Post-accidental Situations. A review of advantages given by modern software like data bases and associated models, geographical information system and remote sensing picture processing is presented. Examples of application both in France (Emergency Technical Center) and in Russia (Institute of Nuclear Safety) is given.

2. Software tools

2.1. Environmental Decision Support System (EDSS)

For emergency situation following a nuclear accident, EDSS must include at least:
-databases concerning the environment: roads, cities, demography, rivers, geology, hydrogeology, agriculture, cattle, food production, and all economic informations needed for the description of region of interest (in a radius of 100 km or more)
-mathematical codes for modelling the transport of radioactivity in air (atmospheric dispersion codes), soils (migration codes), water (river diffusion codes) and food chains
-user friendly interfaces necessary for operating these complex tools.
EDSS can be extended to connect other topics like dose-effect models, or risk analysis.

2.2. Geographical Information System (GIS)

Modern GIS technology is organized as a collection of hardware, software, geographic data designed for efficient capture, store, update, manipulate, analyze, and display all forms of geographically referenced information [2-4]. Thus, GIS includes graphic programs with a data management system (relational data base management systems, as a rule). In this case it is possible to prepare and manage discrete layers of maps along with descriptive data attributes. Map layers consist of points, lines, and polygons and form topological structure of maps. One of the main advantage of a GIS is that it allows one to define the spatial relationships between map features and it holds data bases [2]. The simplest "GIS" can be organized as a graphic data base management system of spatially (geographically) related data.
There are two widely used types of picture in infography: vector based, and raster based [3].
In raster based systems, the picture is a combination of sets of colored points (picture elements or "pixel"), similar to a photography. In vector-based system, each line element is defined with two points (vector) and the result is a drawing, defined with points, lines and polylines of variable thickness. Raster pictures are relatively easy to acquire (videocamera), and can be displayed with short time. Major inconvenients are that these pictures can not be modified, updated, and overlaid without the help of sophisticated processing softwares. They need high storage content.At the opposite, the advantages of vector picture are those of vector: rotation, translation, etc. Overlaying is easy to perform, and need few storage content. Inconvenients are difficulties encountered for data acquisition directly or indirectly by extracting vector pictures from raster pictures.
So, both DSS/EDSS and GIS we are going to use depend on many factors (decision to be taken, speed, efficiency, flexibility and compatibility, data base management, hardware, etc), that should be taken into account for the DOM. In order to satisfy sometimes contradictory objectives different combinations of DSS and GIS possibilities should be used for final presenta-

tion of the results and analysis.

2.3 Image Processing

One of the main problem which arises when making pollution assessment, is to determine the extension and the level of contaminated biomass (agricultural field, forest, etc). These lacking informations, concerning land cover may be obtained with satellite image. Advantages are the "natural" synchronism and easy updating of informations and disavantages are the necessity of processing for making data comprehensive.
Information given by satellite picture, like SPOT, or LANDSAT TM, is the capacity of soils and plant to reflect more or less solar radiation. Under certain circumstances, vegetation can be identified by its "spectral signature", especially during high chlorophylian activity. This identification is enhanced by studying combination of two detection channels like red and near infra-red (normalized vegetation index).
Processing the whole picture allows the determination of "contour" zone -area of same use. At this step, the picture may be transformed from raster to vector mode and be exported towards a G.I.S, each contour beeing linked with a type of vegetation, and corresponding data like interception factor of radioactivity by leaves, or agricultural production rate, both depending on cultural calendar [5].

3. Use in Emergency Center

Two illustrations of software tools used in Emergency Center are given, one concerning more the geographical mapping at the Emergency Technical Center of IPSN (France), the second more oriented on EDSS, and especially on analysis of the consequences of Chernobyl accident at the Institute of Nuclear Safety (Russia).

3.1. French Emergency Center

In France, in case of a severe accident on a nuclear facility, IPSN Emergency Technical Center (ETC) must provide technical assistance to the different Ministry Departments and more particularly to the Direction for Safety of Nuclear Installation (DSIN).
During the early phase of the accident, the goal of ETC is to evaluate the accidental situation and to forecast its development in terms of release to the environment. The ETC is organized round a management unit receiving data from two working parts:
-one is studying the situation within the damaged plant (plant assesment cell);
-the other one is concerned with assessing the radiological consequences of the accident (radiological consequence cell).
After the early phase the second cell is transformed into a post accident one. Radiological assesment cell takes into account the source term and meteorological data to calculate atmospheric dispersion. Two models are runnig on VAX computer: a plume one and a more sophisticated software that is a puff

model. These models compute estimated values such as deposits, doses and dose rates, and share out them on a spatially two dimentional regular grid. Then a specific software extracts contour pattern from the base grid predicted values. At this step , cartography system is used. The aim of this latter is to provide decision-oriented maps with enough information for the concerned authorities.
The major functions performed by the ETC cartography system are: 1) Display isovalues obtained with the atmospheric dispersion models; 2). Add other topics, to be overlaid on the preceding map: geographical maps, collected measurements in the environment, demography, grid marks -UTM, Lambert; 3). Modify the scale; 4). Evaluate demography or any other parameter inside a defined area with the use of mouse and screen.
Because of the accuracy and level of detail required, digitized data are coming from different scales of map. Four classes are defined: site scale, local scale, regional scale and national scale. But it must be pointed out that despite the discontinuous origin of data, all topics can be displayed on a continuous way, as the user wants. Coordinates used are either french (Lambert) or international (Universal Transverse Mercator).
This system being used for emergency situation, a great effort was made for improving the speed of the software: demography evaluation needs only 3 seconds. For portability and maintability reasons, the software is written in normalized languages -FORTRAN 77 and GKS (Graphical Kernel System)- running on mini computer VAX Station 2000 and Micro computers.

3.2. Russian Emergency Center

Recently in Institute of Nuclear Safety (IBRAE), a comprehensive integrated system "RCHAINS" was deleloped for modelling of radionuclide behaviour in environment. It is organized as EDSS, and includes a number models of different complexity (modelbank) for modelling of atmospheric dispersion, migration in soil, food chains, dose-effect models, risk analysis, etc.
All models consist of processors, model-dependant data bases (coefficients, parameters, spatially/geographically distributed data from GIS, etc) and user-friendly interfaces. Relational data bases are also an input and output of modelling and perform integration.
Such approach is called "subject oriented one". It is highly flexible and efficient and can be used both in emergency and post accident situations. Besides geographical layers different types of models layers are organized as a data bases. Thus, all results of modelling can be presented and analyzed with the help of raster or vector-based systems.
Electronic maps are of different scales and obtained by digitizing of existing ones along with attribute data. The primary layers (hydrography, transport network, settlement, land use, etc) are selected for preparing maps for the decision analysis with DSS: different results of modelling (data bases), and data bases from in situ measurements and monitoring systems are processed for presentation on different maps.
Up to four different data bases can be presented, analysed, intercompared simultaneously on the same monitor. The whole

system is organized as a local network of PC compatible computers and several experts can work on the same subject. For the rapid management raster-based maps also can be used. In the latter case for editing point objects specially developed "EDITOBJ" module for moving pollution sources, analysis of countermeasures like relocation is widely used.
At present time, the main activity concerns the situation after the Chernobyl accident. This work is organized within the framework of Information Analytical Centre of IBRAE [6]. An information about radiological, socio-economic and other situations is collected and analyzed on thousands of settlements. These comprehensive data banks are used for expert analysis, prognoses, and preparing different types of maps for the decision makers. In this case, the main tasks are connected with powerful DBMS of geographically related data. Different scales (from tenth of meters to hundred kilometers) and usually randomly distributed data require corresponding software modules for data processing that includes statistics, geostatistics, interpolations, fractal dimension analysis.

4. Conclusion

Making decision in emergency or post accident situations after a nuclear accident is a difficult task, which can not be performed without using a panel of modern tools like Environmental Decision Support System, Geographical Information System, Satellite Image Processing. The complexity of the problem corresponds to the sophistication of these tools, which can not exist without software frame.
At present, the main unsolved problem is to connect all these softwares together for making them compatible, particularly from the point of view of files tranfer (import and export).

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THE MONITORING ORGANISATION OF THE SWISS *EMERGENCY ORGANISATION RADIOACTIVITY*

Christian Wernli[1], Dominique Rauber[2], Serge Prêtre[3], Ueli Niederer[3]

[1] Paul Scherrer Institut (PSI), Villigen; [2] Nationale Alarmzentrale (NAZ), Zürich; [3] Hauptabteilung für die Sicherheit der Kernanlagen (HSK), Würenlingen.

Introduction

The Swiss Emergency Organisation Radioactivity (EOR) consists of:

- the Radiation Emergency Management Board (Leitender Ausschuss Radioaktivität, LAR) being responsible for taking appropriate actions in case of an emergency
- the National Emergency Operations Center (Nationale Alarmzentrale, NAZ), the permanently operated station for first-stage actions, collection and evaluation of monitoring data, and proposal of preventive and protective actions
- the Radiological Monitoring Organisation
- the Communication Networks
- further supporting Organisations

Detailed information on the EOR is given in the paper by H.H. Brunner [1]. Until 1991 external and internal irradiations were dealt with by different groups. In these groups not only monitoring but also evaluation, interpretation and recommendation of measures were studied and prepared. Therefore, the existing documentation is separated in applications for external and internal irradiation [2,3]. In 1992 the organisation was changed into a section which is responsible for the whole monitoring systems of both types of irradiation (the Monitoring Organisation) and a second section which is responsible for data evaluation and elaboration of measures to be taken. The tasks of both sections overlap at the data information system (called "Phönix"). The first section is feeding the system, the second is working with their outputs. This new organisational structure became effective at the beginning of 1992.

Structure of the Monitoring Organisation

The Monitoring Organisation can be divided into 4 subgroups (figure 1):

- Networks
- Laboratories
- Instruments and organisations for mobile monitoring
- Data Information System (Phönix)

Most elements of this organisation are in routine operation for various institutions and are put together in the Monitoring Organisation of EOR for training courses and in case of emergencies.

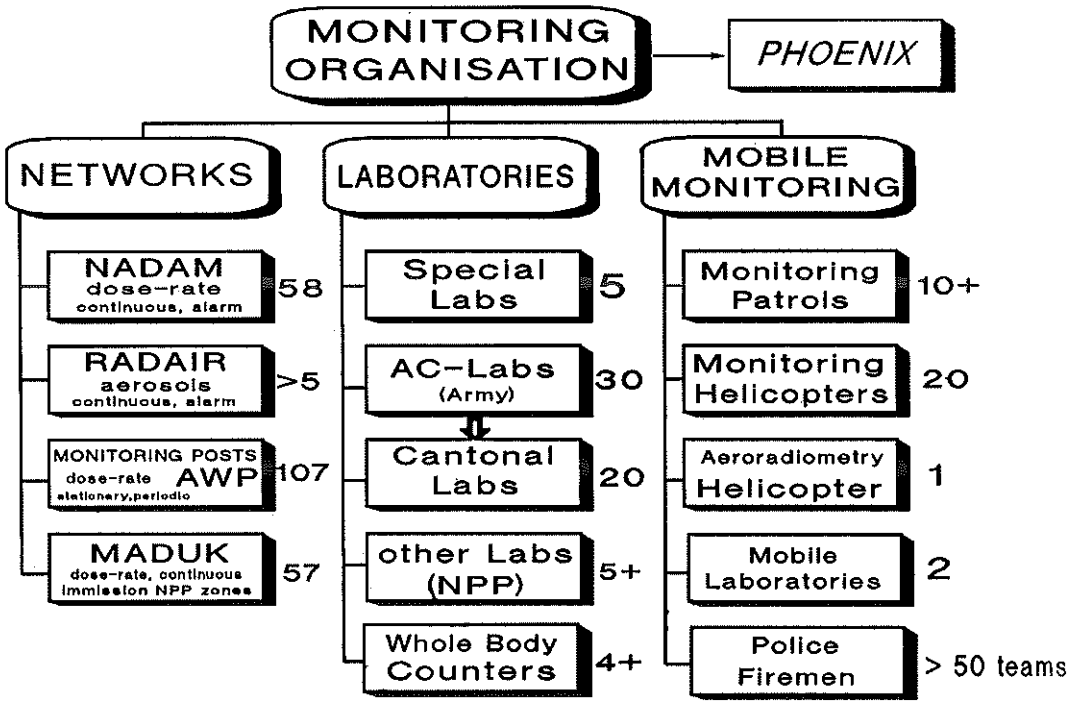

Figure 1: The Monitoring Organisation of EOR

Networks

NADAM

The nationwide photon dose rate monitoring and alarming system is called NADAM. In total 58 stations are distributed all over the country. The readings in the range of 50 nSv/h to 50 mSv/h are transmitted every 10 minutes to the National Emergency Operations Center (NAZ) in Zürich where the data are checked for consistency and included in the Phönix system. The validated data of 16 stations are transferred to TELETEXT and can be recalled on properly equipped TV receivers. In addition, all data are published weekly in the Bulletin of the Federal Office of Health. The positions of the NADAM stations are shown in figure 2. Further Information on the NADAM system is given in the paper of D. Sulmoni [4].

Monitoring Posts (Atomwarnposten, AWP)

107 Monitoring Posts are installed mainly at police stations in cities and villages. These posts are equipped with 2 dose rate meters covering the range of 50 nSv/h to 10 mSv/h and 10 µSv/h to 10 Sv/h respectively. The posts can be alarmed by the NAZ within a few minutes. The personnel are trained to use the instruments and to perform different measurement programs. The results are reported to the NAZ. The local distribution of the posts is given in figure 3.

Early Warning System FWP and RADAIR

A network of 9 continuously working air samplers with beta counters and alarming units has been in operation for more than 20 years (see figure 4). The replacement of these instruments by the so called RADAIR system operated by the Federal Office of Health (BAG) is being planned. This project includes a network of air samplers with automatic alpha/beta counting and gamma

spectrometry units as well as iodine monitors. The new network should be realized between 1992 and 1996.

MADUK
The MADUK system is a planned on line dose rate monitoring network with 57 stations around the 4 swiss nuclear power plants. MADUK will be operated by the Nuclear Safety Division (HSK). The readings are also transmitted to the NAZ and included in the Phönix system. It is planned to have MADUK realized by the end of 1993. Around each of the 4 NPP shown in figure 5 a network of 12 to 17 sensors at distances up to 5 km will be installed. More details on MADUK are given in the paper of F. Cartier [5].

Laboratories

For radionuclide identification and for activity measurements the Monitoring Organisation includes a number of laboratories from different organisations:

Special Labs
Five permanently operated radioactivity measurement laboratories are in the position of so called "Special Labs". These laboratories have a pre-defined environmental measurement program to start with immediately after a request by the NAZ. This first program emphasizes the identification of the nuclides involved and measuring environmental samples. Later on these labs perform measurements according to the program continuously developed by the NAZ. The laboratories are equipped to perform qualitative and quantitative determinations of alpha-, beta-, and gamma emitting radionuclides in various samples, especially foodstuffs. The special labs report their results to the NAZ by Fax or later on directly into Phönix. The position of the special labs are given in figure 6.

AC - Laboratories of the Army
The army is operating 30 AC labs equipped with NaI detectors for gamma spectroscopy. In case of an emergency these laboratories have to be activated and the staff has to be called for military service. Therefore, these labs become effective only several days after activation. Due to their high capacity, they play an important role in the so called main and late phase of an emergency situation. The measurement programs of these laboratories are determined by the NAZ and the results are reported there by Fax. The positions of the AC-labs are shown in figure 6.

Cantonal Laboratories
The new regulations state that the cantons are responsible for the control of radioactivity in foodstuffs. Therefore, several cantons operate well equipped radioactivity measurement laboratories permanently (figure 6). Some of them are also involved in routine environmental surveillance. In case of emergency there is a close cooperation between the AC laboratories of the army and cantonal laboratories.

Other Laboratories
Besides the official laboratories of governmental organisations a number of private organisations operate modern radiation measurement laboratories. These are mainly the nuclear power plants and some industries. Cooperation of these labs with the Monitoring Organisation is informal and deserves a special agreement in case of need.

Whole Body Counters
Whole body counters are in routine operation at PSI and at two hospitals in Basel and Geneva. No decision is taken so far, whether a new whole body counter will be built at Inselspital in Bern. At the moment, the instrument of PSI only is part of the Monitoring Organisation. It is planned to include the whole body counters of the hospitals and to purchase two mobile units for in vivo measurements (see also Mobile Monitoring).

Mobile Monitoring

Monitoring Patrols
Ten monitoring patrols from different organisations are equipped with instruments for dose rate and contamination measurements as well as equipement for air sampling and collecting environmental probes. Around the nuclear power plants monitoring routes are predefined to allow very quick actions after suspected incidents (figure 7).

Monitoring Helicopters
Twenty units of dose rate monitoring systems are available for use in helicopters of the army. One monitoring helicopter is ready for operation all the time. The others would have to be equipped first.

Aeroradiometry
One 16 liter NaI spectrometry system is ready to be used for aeroradiometric measurements out of an airplane or a helicopter. The main use of the system is in searching for distributed sources, e.g. debris of a satellite with nuclear power systems.

Mobile In Vivo Counting Systems and Measuring Labs
It is planned to build 2 mobile laboratories comprising thyroid, thorax and eventually whole body measurement instruments as well as a small laboratory unit for gamma spectroscopy, dose rate, and contamination measurements. The realisation should be completed by end of 1993.

Police, Firemen
Around 50 teams all over the country are instructed and equipped for dose rate and contamination measurements. These teams are responsible for first measurements and actions in case of local accidents involving radioactive substances, such as a fire or a transport accident.

AC Officers of the Army
In case of an accident in a Swiss nuclear power plant, up to 24 dose rate monitoring teams of specialists of the army can be activated for measurements at predefined locations or on routes defined by the so called "Messleitstelle" in cooperation with the NAZ.

Army and Civil Defence Organisation
The army and the civil defence organisation possess 25'000 dose rate monitors each. These instruments with ranges of 10 μSv/h to 10 Sv/h are for use of both organisations themselves.

Data Information System (Phönix)

In an emergency all data from the various networks, laboratories and mobile monitoring teams are centrally collected, stored and evaluated at the National Emergency Operations Centre. The data information system Phönix is being installed for a timely processing of all the products required for an evaluation of the situation.

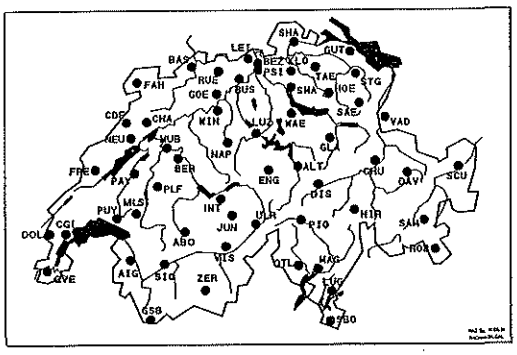

Figure 2: The NADAM network of 58 dose rate monitors.

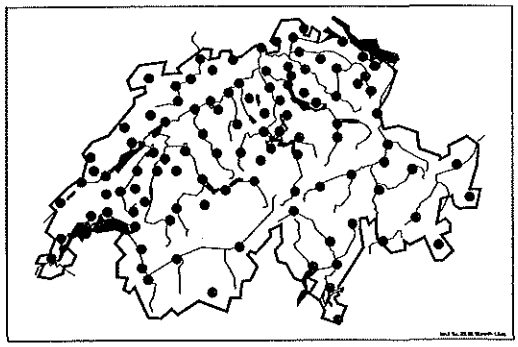

Figure 3: The network of 107 monitoring postes (AWP).

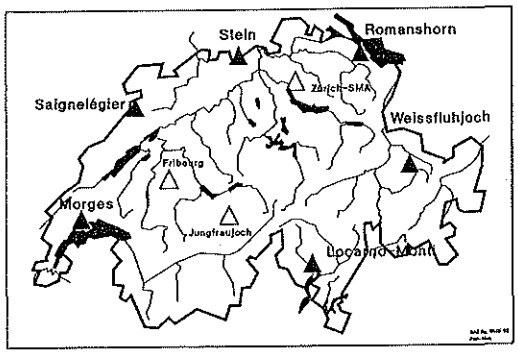

Figure 4: The network of 9 air monitors (FWP).

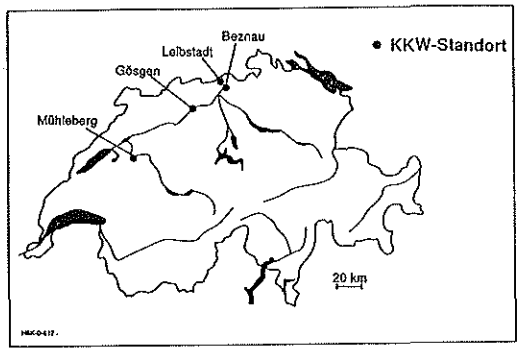

Figure 5: Localisation of the MADUK networks. 12 to 17 dose rate monitors are located within 5 km around each of the four nuclear power plants.

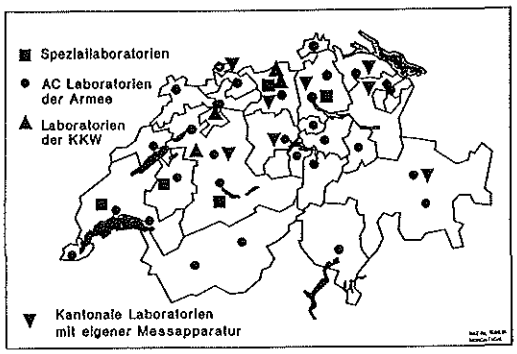

Figure 6: Localisation of the radioactivity measuring laboratories.

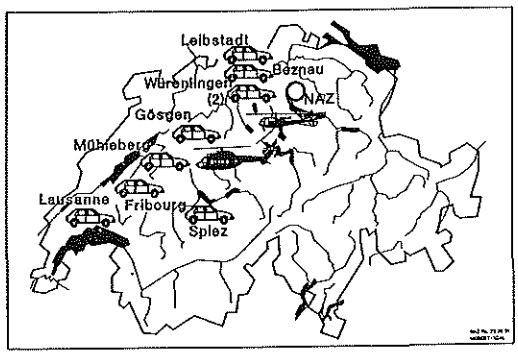

Figure 7: The monitoring patrols and helicopters of different organisations.

In a first step the laboratories are connected to the system. A data base for the laboratory data has already been created and will be used by all laboratories when the system will be completed.

In a second step the evaluation part of the system will be set up. The data of the networks and of the mobile monitoring teams will be combined and evaluated together. The data will be treated to such an extend that the required graphs and overviews can be produced and input data for forecast models become available. The graphical presentation of the data is part of an Advanced Relational Geographic Information System (ARGIS).

Training and Quality Assurance

Most of the instruments, systems and laboratories of the Monitoring Organisation are in permanent routine use. Here quality assurance is performed on a professional level, including training courses and intercomparison exercises. The extension of the organisation in case of an emergency is realised by military persons and equipment. Quality assurance and training of personnel of this part of the organisation is covered by regular military repetition courses.

Ongoing Studies

The Monitoring Organisation must in the near future solve the problems of individual dosimetry of members of the EOR possibly being exposed to radiation, and contamination monitoring of members of the EOR, as well as of members of the population. Both problems are being studied by working groups.

Acknowledgement

The authors wish to thank their collegue Ueli Niederer, the former leader of the internal dosimetry group, who died on March 30, 1991, for his excellent work to the benefit of the Monitoring Organisation.

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DECISION SUPPORT SYSTEM FOR EVALUATING COUNTERMEASURES TO REDUCE INGESTION DOSE

Andrea G. M. Schenker-Wicki; Dominique Rauber
National Emergency and Operation Center, P.O. Box, 8044 Zurich, Switzerland

Summary

In the last twenty years technological changes have taken place very rapidly. Unfortunately, these changes had not only had advantages but also disadvantages, espcially with respect to envrionmental issues. Due to the short- and long-time consequences environmental accidents can trigger and the stress situation arising for companies and gouvernments concerned, the need for a suitable decision support has been realized by the responsible people.

To help overcome the typical difficulties that can arise in a crisis situation, especially in the case of an accidental release of radioactivity, the National Emergency Operations Center has designed and implemented a Decision Support System DSS to evaluate acceptable countermeasures to reduce ingestion dose. The system involves all the necessary modules and techniques for efficient decision making, based on the most recent developments in decision theory, as well as the necessary structuring of the decision making process.

1. Introduction

The Decision Support System DSS proposed in this paper (fig. 1) is especially suited to the political conditions in Switzerland, but could easily be adapted to the conditions in other countries.

The decision power is divided among experts and politicians. The decision process itself starts at the expert level, which is a more technical level. There the different countermeasures to reduce ingestion dose are discussed and checked with respect to technical feasibility and efficiency. In a second step the countermeasures proposed by the technicians have to be judged with respect to political criteria. Finally, the most effective countermeasures with the best political acceptance are proposed to be implemented.

To guarantee decisions of high quality, as many countermeasures to reduce ingestion dose as possible have to be presented to the decision makers. Due to the difficulty of evaluating all the possible countermeasures the space of alternatives is automatically generated. This prevents the decision makers from information overload and is possible by filtering out the irrelevant data through different generic restrictive systems. The countermeasures for reducing the ingestion dose involve such factors as different processing techniques in agriculture or food industry.

In the next paragraph a short overview of the system will be given. The whole system is highly interactive to guarantee the flexibility which is necessary to cope with crisis situations.

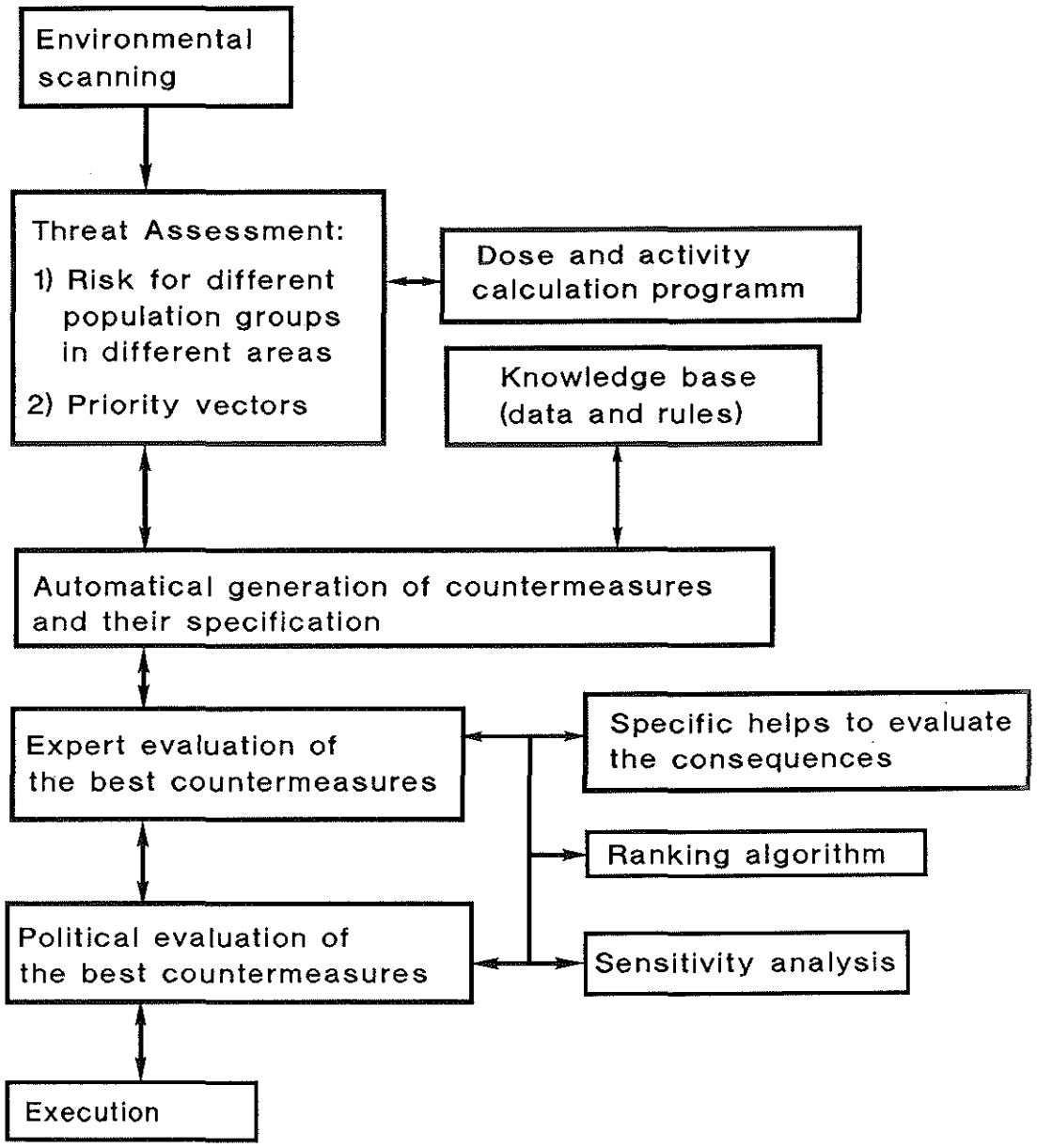

Fig. 1: A Decision Support System for Evaluating Acceptable Countermeasures to Reduce Ingestion Dose

2. Threat Assessment

In a first step the threat for the population concerned has to be assessed. Since radiation damage resulting from ingestion of contaminated foodstuff cannot be determined directly, several calculations have to be made before being able to assess the different kinds of risk thoroughly. These calculations involve the evaluation of the activity concentration in the contaminated foodstuffs and the committed dose equivalent. The calculation code used in the DSS is the German code ECOSYS (1).

Because deposition of radioactivity is unlikely to be homogeneous, the entire territory of Switzerland is divided into different areas: regions with different levels of deposition and typical agricultural structures are distinguished. The size of each area can vary; the smallest units are political districts. Based on these data activity concentration in foodstuffs and the resulting ingestion dose for the population in the corresponding area can be calculated. In order to guarantee the highest possible reduction of the ingestion dose, countermeasures for the product with the highest relative dose contribution in the first year after the accidental release of radioactivity are evaluated as a priority.

3. Generation of the Space of Countermeasures

A large number of countermeasures to reduce the ingestion dose have been evaluated in a literature study (2). However, since some alternatives are not feasible at a certain time because of physical, chemical, biological or time restrictions, the list of evaluated alternatives has to pass a filtering system. Here those alternatives which do not comply with the restrictions are eliminated.

After having evaluated the single countermeasures allowing the reduction of nuclide concentration in soils, crops, animals and foodstuffs the decision makers have to determine how these activities can be combined in a most efficient manner. The system gives all the combining facilities and allows its users to cope with more than 2000 countermeasures for different kinds of foodstuff.

4. Criteria and Method Used for Decision Making

In many cases the countermeasures are difficult to compare because of the fuzzy set of consequences their implementation can induce to the environment considered. To support decision making efficiently the badly defined consequences must be systematically investigated and explored. To cope with this demand a coherent family of criteria was evaluated (3):

Technical feasibility:
The technical feasibility deals with technological problems mainly concerning changes in the production processes, such as additional capacity with respect to room, storage place, machines, running materials or qualified staff etc.

Problems of radioactive waste management:
To judge the different alternatives the entire inventory of radioactive waste arising within the period during which a countermeasure is executed, has to be estimated by the technical experts.

Reduction of food-supply:
Reduction of food-supply or rationing of foodstuffs could cause disturbances and panic among the population. Therefore every negative deviation from the average food or feeding supply-level should be prevented.

Estimation of costs:
To estimate the costs, the additional expenses for a certain countermeasure have to be summed up. Because a part of the costs can sometimes be transferred to insurance-companies the different possibilities of cost apportion should also be considered for decision making.

Consistency of national decisions (countermeasures) with international ones:
If the countermeasures taken in Switzerland differ too strongly from those countermeasures implemented in other countries with a similar deposition of radioactive material, the population begins to doubt about their accuracy and efficiency. A gap between national and international countermeasures should therefore be avoided.

Acceptancy:
The expected change in the consumption behaviour is also a consequence which should be taken into account for final decision making. If the change in the behaviour is expected to be very small, the implementation of a certain countermeasure will be much easier and more successful.

Theoretically possible dose reduction:
To calculate the individual dose reduction, the model ECOSYS (1) is used.

If multiple criteria have to be considered for decision making, more than a hundred methods have been developed over the past few years to aggregate the different criteria. The approach implemented in the DSS belongs to the multi-attribute projection methods (4). This set of methods is mainly considered for the treatment of qualitative data. Due to the lack of information and the stress arising in a crisis situation a complete and perfect modelling of the decision makers' preference system is assumed to be unrealistic.

5. Evaluation of the Best Countermeasures

After having determined the feasible countermeasures, the experts have to decide how reasonable they are, largely based on different technological criteria. The most reasonable countermeasures have to be evaluated separately for each product. Only these will then be proposed to the politicians to test their acceptance with respect to political and economical criteria.

To avoid unnecessary discussions, the politicians should only be confronted with efficient and feasible solutions, from a technological point of view. The politicians are free to accept the rankings of the experts without any change, to accept the list of propositions, but change the rankings or to return the proposed solutions asking for different ones.

The consequences of the different countermeasures with respect to a certain criterion are evaluated through special models. These models involve both, quanitative approaches as well as qualitative ones.

6. Conclusion

The concept designed in this paper is a framework to help the decision makers in complex decision making situations. If the system is to be used for decision making in a crisis situation, it must be dynamic, flexible and adaptable to the ever changing outside parameters. The need to gain organizational acceptance is another important premise. To gain this acceptance the responsible authorities and experts have been involved in the planning and implementation process. A first prototype version of the

system is now being implemented at the National Emergency Operations Center and will be tested with technical and political decision makers in Switzerland.

7. References

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INFLUENCE OF THE CONTAINMENT VENTING SYSTEM ON THE EMERGENCY PLANNING

M.Baggenstos, Swiss Federal Nuclear Safety Inspectorate
CH-5232 Villigen-HSK

Dr. B. Covelli, TECOVA Litzibuech 8
CH-5610 Wohlen

Summary

The main purpose of the containment filtered venting system is to prevent an overpressure failure of the containment with an uncontrolled release of radioactivity.

The paper discusses the existing practice for emergency response: recommendation of prophylactic sheltering as protection against the exposure from the cloud. The area involved and the timing for this countermeasure is determined by the accident progression and the wind direction at the time of release.

With a controlled filtered containment venting the time of release and therefore the involved area (and population) can be influenced. With such a system an optimisation of the countermeasures can be achieved.

The venting strategy includes the following elements:

- containment pressure
- containment H_2 content
- containment steam content
- containment radioactivity content
- weather category
- wind direction
- status of the public protection

Optimisation of the initiation of the venting system is discussed, using the programm VENTOPT.

1.Emergency Planning : Existing Practices

The existing emergency planning practice in Switzerland is described in the federal concept " Emergency Planning and Preparedness for the Vicinity of the Nuclear Power Plants " published 1991.

1.1 Emergency Planning Zones and Sectors

The vicinity of the NPPs in Switzerland is divided into 3 zones.

Zone 1 (3-5 km) comprises the area around a nuclear power plant in which immediate protective measures are required.

Zone 2 (20 km) adjoins Zone 1 enclosing an area with an outer radius of approximately 20 km . Zone 2 is divided into 6 sectors each of 120°.

Zone 1 and Zone 2 are the plume exposure pathway Zones

Zone 3 contains the area outside 20 km. Measures to protect the public during the passage of the radioactive cloud are not necessary. Preplanning are arranged for the ingestion pathway.

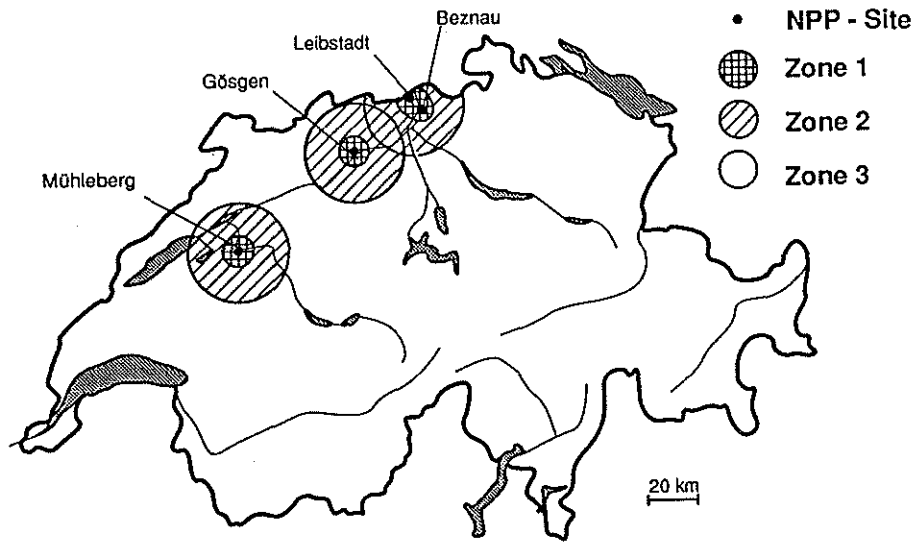


Fig. 1 Emergency Planning Zones in Switzerland

In the case of the NPP-Sites Beznau and Leibstadt, the Zones 1 and 2 are combined

Special Notes

- Zone 2 is divided into 6 overlaped sector
- Warning is always give to complete Zone 1 and complete Zone 2
- Alert ar always give to the complete Zone 1 and also to endangered sectors of Zone 2

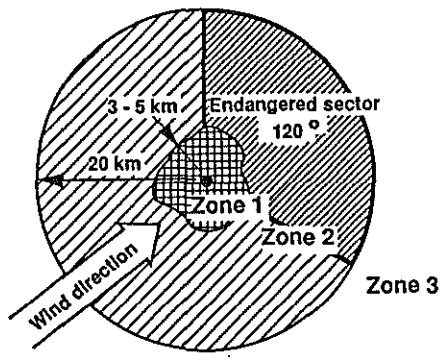


Fig. 2 An endangered sector in Zone 2

1.2 Alert Procedures

If an accident poses a threat to the environment, a three stage warning and alert procedure is set in motion. The criteria for issuing warnings and alerts are given in the Emergency Regulations of the NPP.

A WARNING is issued when an accident occurs in a NPP, but there is no immediate threat to the environment. The WARNING puts the federal, cantonal and community organizations on stand-by for a possible alert.

A GENERAL ALERT is issued when an accident evolves in such a way, that it could lead to a danger to the environment. The population at risk is alerted to initiate preparations for a possible release at the NPP.

The RADIATION ALERT is given when the release of radioactive material has begun. The population is instructed to take immediate shelter in a cellar or civil defense shelter.

The alert procedures is dictated by the situation in the plant. The release timing cannot be chosen therefore also the weather situation and the endangered population is dictated by the accident progression in the plant.

It is possible, that the release starts during a (radiologicaly) bad weather situation and in a wind direction with a high population density.

2 Influence of the Filtered Venting System on the Emergency Planning

With the filtered venting system for some scenarios the timing of the release (and therefore the weather situation) can be influenced by the activation of the system.

2.1 Influence on Emergency Planning Zones

The EPZ are not influenced by the system because the main contribution to the external doses is determined by the radioactive noble gases which cannot be retained in the filter system.

2.2 Influence on the Alert Procedures

The decision to activate the filtered venting system can be chosen within some time interval. The elements which are taken into account for this decision are listed in Fig 3.

2.3 Optimisation of the Filtered Venting System Activation with the VENTOPT Program

The PC-program VENTOPT calculates the release of radioactivity and transport to the environment in the course of a core melt accident upon activation of the filtered venting system (FVS) . The pressure build-up in the containment and the radioactivity content in the containment are taken from risk studies. It is also possible, to introduce actual data from PAS-measurements (Post Accident Sampling).

The VENTOPT code has 3 parts:

CHROCO: calculates the actual activity contents in the containment

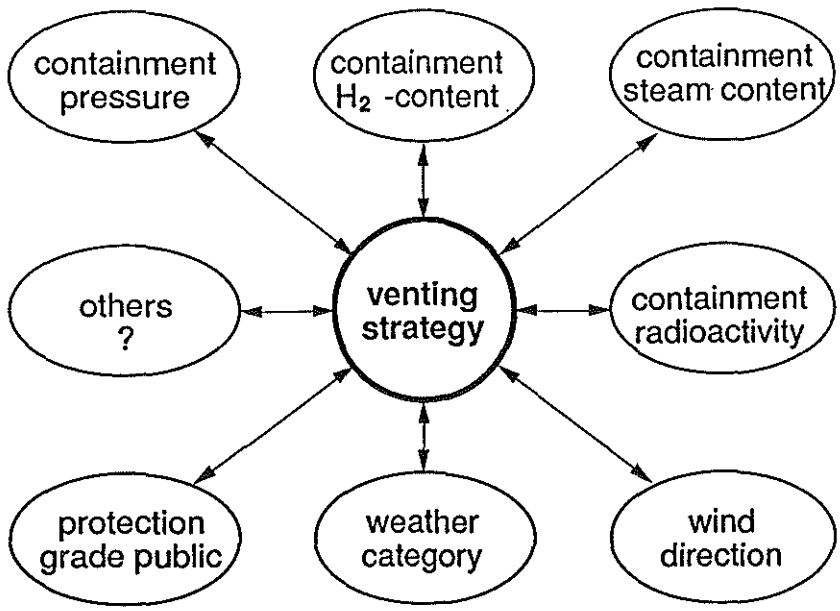
CHROIN: organizes the input files for the transport calculation

CHRONY: calculates the off-site doses

The population density is integrated in the programm, so that collective doses can be calculated.

Within the context of this paper,the program is only used to compare different venting strategies and not to calculate absolute values .

Figure 3: Elements of the venting strategy



2.3.1 Calculation of the Released Activity after Different Times

(Fig 4; Tab 1)

The released activity is influenced by

- radioactive decay
- sedimentation inside the containment
- resuspension and revaporisation from the pool

The iodine source term (st) decreases a factor of ten within 4 hours. But within the next day it decrease only slowly (a factor of 2).

The aerosol st has a minimum after about 10 h.

2.3.2 Influence of the Weather Situation on the Involved Persons and the Area

(Fig 5; Fig 6)

It is well known that the weather situation greatly influences the radiation doses. It is not useful, to compare the dose at a given time for different weather situation, because at the time of a accident, the weather situation cannot be influenced. We have therefore taken a typical weather situation during a winter day (as example near the NPP gösgen) and (assuming, that these situation can be prognosed for some time) have calculated the influence of the weather changes within this time on the dose.

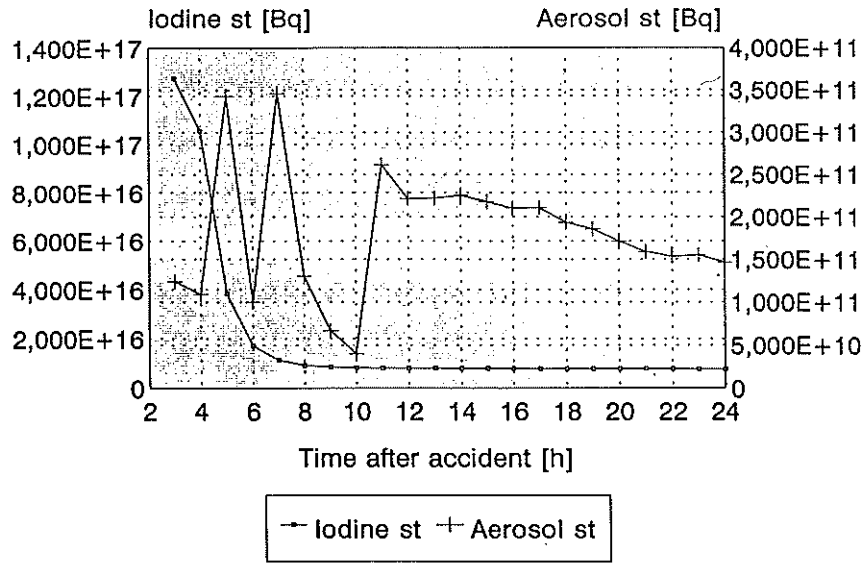
The cloudshine doses are below the limit (0.01 Sv) if the system starts between 9 and 14 h . The thyroid doses are below the limit (0.03 Sv) between 11 -14h and 17-19 h.

The contaminated area is minimal when the system is started between 19 and 20 h. (Fig 6)

Conclusion

With the filtered venting system, the affected area and therefore the involved persons (and their doses) will be influenced greatly by the time when the system is activated. It is not optimal, to wait until the rupture disk of the system will open and start the release. The decision to start the FVS gives the involved emergency staff an large (and new) responsibility. The VENTOPT program helps to take this decision.

Fig. 4: Source term (st)



Tab. 1: Weather condition

Start venting	Weather condition
3	C,230,0.3,t,1000
4	C,230,0.3,t,1000
5	C,260,0.4,r,1000
6	C,260,0.4,r,1000
7	C,280,0.3,r,1000
8	C,280,0.3,r,1000
9	C,240,2.5,rr,1000
10	C,240,2.5,rr,1000
11	C,240,2.4,rr,1000
12	C,240,2.4,rr,1000
13	D,250,2.0,r,1000
14	D,250,2.0,r,1000
15	D,270,1.5,r,1000
16	D,270,1.5,r,1000
17	F,280,1.0,r,1000
18	F,280,1.0,r,1000
19	F,270,0.5,t,1000
20	F,270,0.5,t,1000
21	D,280,0.3,t,1000
22	D,280,0.3,t,1000
23	C,280,0.2,t,1000
24	C,280,0.2,t,1000

Meteorological legend: D,280,5.3,t,1000
D=Stability condition
280=Winddirection 5.3=Windspeed
t=dry r=weak rain rr=heavy rain
1000=Height of inversion

Fig. 5: Involved Area around 20 km of the plant
Meteorological condition: Winter

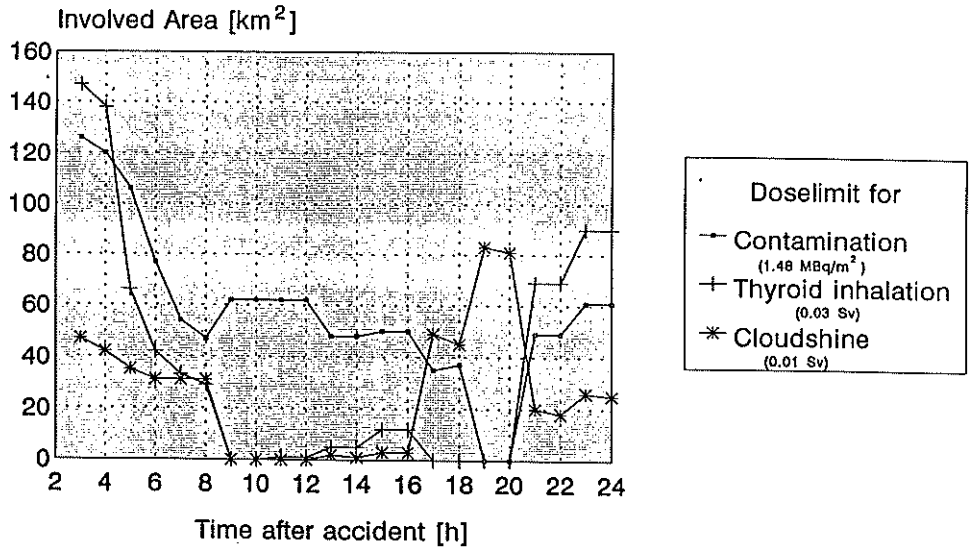
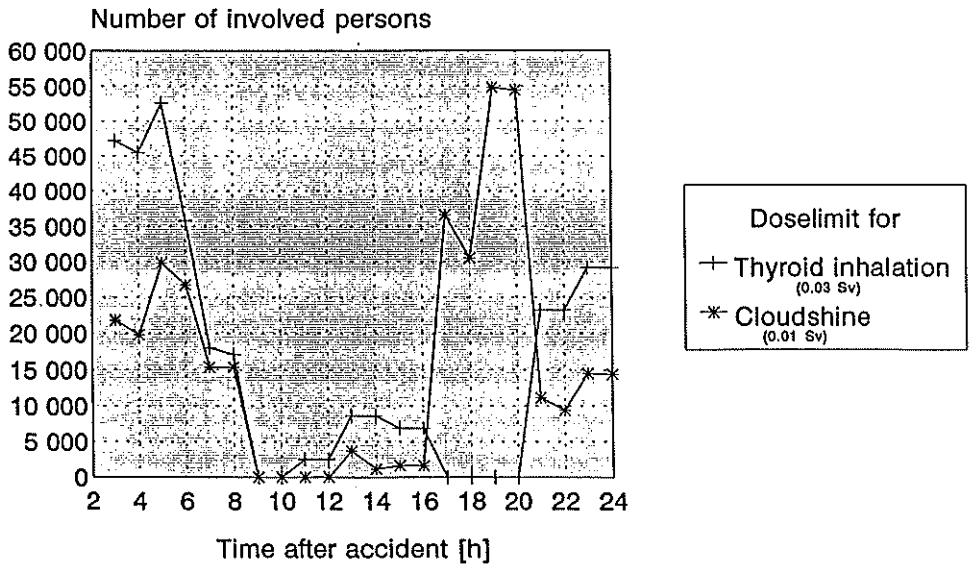


Fig. 6: Number of involved persons around 20 km of the plant
Meteorological condition: Winter



PRELIMINARY RESULTS ON TRANSFER OF RADIONUCLIDE IN SOIL AND CROPS IN THE CHERNOBYL AREA

J.M. Quinault[1], R. Arutyunyan[2], P. Picat[1], S. Gavrilov[2], C. Colle[1] C. Friedli[3]
M. Kanevski[2], V. Kiselev[2], P. Fache[1], H. Maubert[1]

[1] INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE, Département de Protection de l'Environnement et des Installations. IPSN/CEA, 13108 SAINT-PAUL-LEZ-DURANCE, CEDEX, France.
[2] INSTITUTE OF NUCLEAR SAFETY. Bolshaya Tal'skaya, 52, 113191, MOSCOW, Russia.
[3] ECOLE POLYTECHNIQUE FEDERALE DE LAUSANNE, Institut d'éléctrochimie et de radiochimie. EPFL-Ecublens, CH - 1015 LAUSANNE.

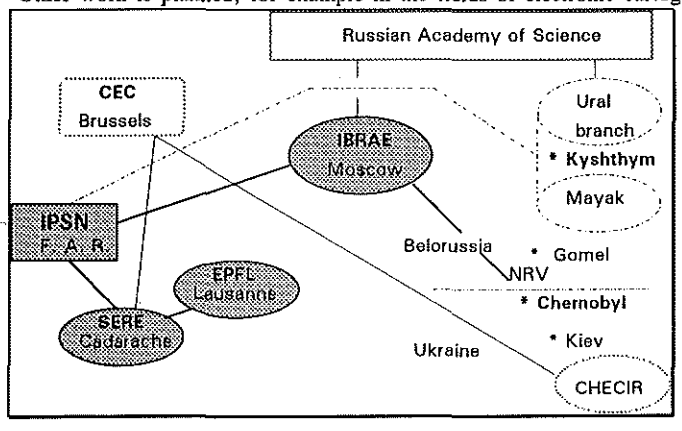
Summary

The IPSN is collaborating with IBRAE (INS: Institute of Nuclear Safety of Russia) in the fields of measurements and computer modeling, and with EPFL (Swiss Federal Institute of Technology, Lausanne) in the field of radiochemistry. The opportunity provided by Russia to collect samples in a contaminated area inside the Chernobyl region enabled us to combine the efforts of the three institutes to provide a complete chain: sampling, measurement, modeling and validation. Emphasis was on the evolution, versus time, of the vertical distribution of cesium in various soils, mainly podzol, measured by gamma spectrometry, and strontium, extracted by radiochemistry and measured by beta counting.
This paper presents the objectives of the project, the progress of the work and the first results obtained.

1. Radioecology collaborations between IPSN, Russia and Ukraine

Russia, Belorussia and Ukraine contain areas that have been highly contaminated by radionuclides from various releases. The French Nuclear Protection and Safety Institute (IPSN) started scientific and technical collaborations several years ago, particularly in the field of radioecology with its "Service d'Etudes et de Recherches sur les transferts dans l'Environnement" (SERE: Environmental transfer research section) of the Department for the Protection of the Environment and Installations (DPEI). For the Urals region studies have been undertaken under the terms of an agreement between IPSN, Mayak (the company responsible for the Kyshtym accident complex), and the Urals division of the Russian Academy of Sciences. For the Chernobyl site IPSN operates under the terms of two agreements : (i) agreement between the CEC (Commission of the European Communities) and CHECIR (CHErnobyl Center for International Research), under Ukranian patronage, for which SERE is concerned by decontamination and rehabilitation strategy, (ii) agreement between IPSN and IBRAE (Institute of Nuclear Safety [INS] of the Russian Academy of Science, Moscow), for whom SERE operates under theme 4, concerning verification of models, cartographic data systems and measurement in accident environments. The present document only deals with work undertaken as part of the last agreement

A collaboration protocol between IPSN and IBRAE was signed in Saint Petersburg on 2 June 1991. This agreement contained various points, including the following concerning radioecology: (i) sampling and measurement of α, β and γ radioactivity in soil and plants, (ii) modeling and validation of the models by comparing them with on-site data in the fields of radionuclide migration in soil and transfer to the food chain. Other work is planned, for example in the fields of electronic cartography and γ spectrometry methods for directly measuring *in situ* contamination.



For many years, IPSN/SERE has also been collaborating in the field of radiochemistry with the Ecole Polytechnique Fédérale de Lausanne (EPFL), Switzerland, particularly as concerns measuring strontium 90 using chemical separation and β measuring techniques. The Lausanne team were thus the obvious choice when SERE needed to take strontium measurements on their Belorussian samples.

Once a definition of the joint work planned had been established (in particular by Messrs. Arutyunyan and Picat), several missions were undertaken in France, Russia and Belorussia by Messrs. Gavrilov and Colle on the samples and measurements, and by Messrs. Kanevski and Fache on the implementation of predictive models. Messrs. Friedli and Colle also met to tackle the problems of using radiochemical techniques for measuring strontium.

2. Sampling, measurement and observations

2.1. Sampling

Soil and plant samples were taken near the village of Dzerjinsk in the Narovlya region (hence the sample-identifying letters NRV) of the Gomel district. The sampling area is located 35 to 40 kilometers North-East of the Chernobyl plant. Although the village was not evacuated after the accident, radionuclide deposit is significant, of the order of one million Becquerels of Caesium 137 per square meter, according to our measurements. Considerable variation is found both in the deposits and the types of ecosystem (natural, semi-natural and cultivated) together with a variety of soils, plants and types of cultivation. 7 cm diameter soil samples were taken to a depth of approximately 20 cm, the standard depth accepted by the UIR (Union Internationale des Radioécologistes). These samples were then fractioned into 1 cm thick sections for uncultivated soils and 2 cm thick for the rest. Plant samples were taken in the immediate proximity of the soil samples. The plants sampled were clover or grasses, depending on what was available at the time of sampling.

Sampling point	Type of soil	Plant
NRV 26	Uncultivated permanent meadow (former garden rich in organic matter)	grasses
NRV 31	Uncultivated permanent meadow	clover + grasses
NRV 34	Cultivated temporary meadow (quasi-uniform radionuclide concentration)	clover

2.2. Measurements and observations

γ measuring on the various samples was performed at Cadarache and strontium measuring was performed in Lausanne. IBRAE undertook gammametry measurements *in situ*.

IPSN performed the laboratory measurements for the principal gamma-emitting radionuclides on all the soil samples brought to Cadarache. The total deposits per m^2 in table 1 were extrapolated from measurements on all the soil samples taken on the various sites. IBRAE used a sensor located 1 meter above ground level, together with a calibration curve to measure the total quantity of cesium 137 deposited. As Table 1 shows, this simple, *in situ* method gave good results, particular where cesium distribution was uniform. In contrast, it underestimated the deposit somewhat (20 to 40%) , when cesium was mainly concentrated on the surface.

Table 1: Total deposit of cesium 137 (kBq/m^2)

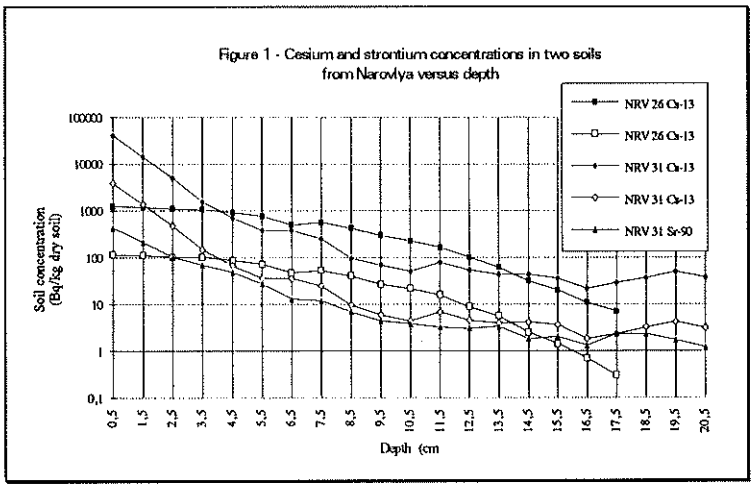
Sampling point	Laboratory measurements [1]	In situ measurements [2]	In situ/lab. ratio
NRV 26	130	80	0.60
NRV 31	850	700	0.80
NRV 34	640	620	0.98

[1] Total vertical profiles measured by IPSN

[2] Overall measurement 1 m from ground level, recorded by IBRAE

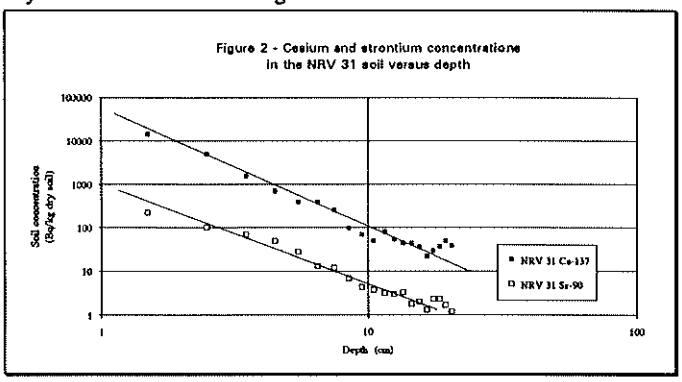
Figure 1 (concentration curves as a function of the depth of the center of each section) shows results of γ measurements for cesium 134 and 137 on the 1 cm soil sections from samples NRV 26 and 31 taken from uncultivated soils after the Chernobyl accident, together with the results for β measurements on strontium 90 at sampling point NRV 31. Given the uniformity of the soil due to the cultivation methods used, soil NRV 34 was not used for studies on the migration of radionuclides.

Concentrations of cesium 134 are approximately ten times



weaker than those of cesium 137 (end October 1991, which is the equivalent of a ratio of 2 in May 1986) to a depth of approximately 12 cm. Beyond this depth, the presence of previously deposited cesium 137 due to fallout from Chernobyl increases this factor of 10. Vertical distribution of cesium 134 and 137 concentrations are parallel and similar, proving that migration is linked to the element cesium and not a function of the 134 or 137 isotope . Although cesium deposition was approximately 7.5 greater at point NRV 31 than point NRV 26, figure 1 shows that the two vertical concentration profiles do not have the same slope and that downward migration was greater in former garden NRV 26, at least to a depth of 14 cm. This is undoubtedly explained by the nature of the soil and the presence of organic matter (peat) that tends to diminish the sorption coefficient of cesium in soil, and consequently to increase downward migration.

Strontium measurements were carried out by C. Friedli in Lausanne. It will be seen that the "strontium-NRV 31" curve in figure 1 has the same general appearance as the "cesium-NRV 31" curve. Figure 2 combines curves for cesium 137 and strontium 90 for sampling point NRV 31 on a log-log graph. These curves have a similar slope. This suggests that the retention and migration behavior of strontium is similar to that of cesium, for NRV 31 soil type.



2.3. Transfer to plant cover

Table 2 shows the results of measurements for cesium on clover sampled at points NRV 31 and 34. The figure also shows the soil-plant transfer factor, in kg dry soil/kg dry plant matter, calculated using the average soil concentration over 20 cm. Although the representative nature of this average is open to question for point NRV 31, where distribution was very uneven, the transfer factors are nevertheless of the same order of magnitude as one another and those given in the literature* .

Table 2 Cesium concentration in soil and clover + associated transfer factor

	NRV 31 non-uniform			NRV 34 uniform		
Radionuclide	Soil[1]	Clover[2]	Ft[3]	Sol[1]	Clover[2]	Ft[3]
^{137}Cs	3,200	210	0.066	2,000	100	0.050
^{134}Cs	300	24	0.080	190	10	0.053

[1]Bq/kg dry soil [2]Bq/kg dry matter [3]kg dry soil/kg dry matter

Table 3 shows results for soil→grass transfer at points NRV 26 and 31. If transfer factors remain within the norms for NRV 31, they should be greater by a factor of 100 for NRV 26. This difference cannot be explained by counting errors, that never exceed 10%. Moreover, the results have also been confirmed by IBRAE, who obtained practically the same values. A tentative explanation of this difference might be in terms of resuspension of radionuclides and their redeposition on plants. This phenomenon may play a part, but further study of the differences is certainly needed.

Table 3 Cesium concentration in soil and grass + associated transfer factor

	NRV 31 uncultivated			NRV 26 old garden		
Radionuclide	Soil[1]	Grass[2]	Ft[3]	Soil[1]	Grass[2]	Ft[3]
^{137}Cs	3,200	220	0.066	425	3,100	7.3
^{134}Cs	300	23	0.076	40	300	7.3

[1]Bq/kg dry soil [2]Bq/kg dry matter [3]kg dry soil/kg dry matter

3. Modeling and validation

A predictive model for the migration of radionuclides deposited on soil, **CATHY**, has been developed by H. Maubert [2] [3]. In brief, the model describes the dispersion of an interactive solution (adsorption-desorption) in a porous medium.

The model (equations and parameters) is established using the following simplifying hypotheses : solution equilibrium between the solid and liquid phases exists and is reached very rapidly, this equilibrium, expressed

* UIR 1987 [1] Soil-to-plant Transfer Factor : Cesium sandy soil Clover 0.06 (dry weight)
Grasses 0.15 (dry weight)

by the sorption coefficient, K_d, is constant in time and space and independent of the total quantity of solution present, the soil is permanently water-saturated, the seepage velocity is constant throughout the observation, the diffusion coefficient of seepage water in the soil is constant and independent of depth.

**Note : The sorption coefficient of soil in place, K_d is dimensionless and is deduced from coefficient K_d', commonly obtained in a closed (batch) reactor, with the dimension *l/kg dry matter* : $K_d = 1{,}33.K_d'$.

$$\theta \cdot \left(\frac{\delta C}{\delta t}\right) + \rho \cdot \left(\frac{\delta S}{\delta t}\right) = D \cdot \theta \cdot \left(\frac{\delta^2 C}{\delta z^2}\right) - u \cdot \theta \cdot \left(\frac{\delta C}{\delta z}\right)$$

$$S = K_d \cdot C$$

C	Radionuclide concentration in the liquid phase	Bq/m^3 water
S	Radionuclide concentration in the solid phase of soil in place	Bq/m^3 soil in place
K_d	Sorption coefficient of soil in place**	dimensionless
θ	Relative humidity of soil: m^3 water per m^3 soil in place	dimensionless
ρ	Density of soil in place	kg dry soil/m^3 soil in place
u	Average rain seepage velocity (permeability)	m/s
D	Diffusion coefficient of seepage water in soil	m^2/s
z	Depth; increases downwards	m
t	Unit of time	s

Validating a predictive model is a difficult operation because various techniques and criteria can be used, which include two distinct objectives (a) testing a standard series of calculations described as a "code", (b) scientific study of the validity of the fundamental hypotheses and the real value of the parameters.

Calculations were performed with standard conditions, i.e. K_d = 500 for cesium and 20 for strontium, total rainfall 600 mm per year and the estimated value of the amount deposited at end April 1986.

Figure 3 - Comparison observed / calculated (standard parameters) concentrations of cesium 137 and strontium 90 in soils from Narovlya

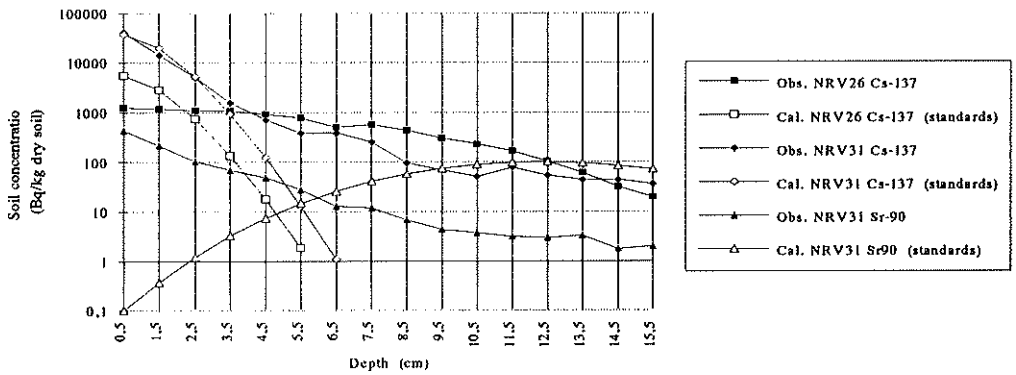
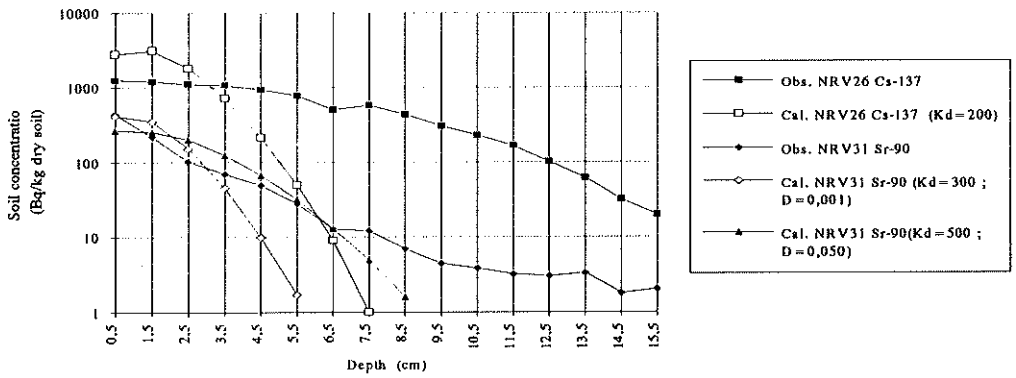


Figure 4 - Comparison observed / calculated (non-standard parameters) concentrations of cesium 137 and strontium 90 in soils from Narovlya



Although our calculation results for standard conditions agreed reasonably well with the observations for cesium, particularly for sampling point NRV 31 for the first few centimeters, the situation was quite different for strontium. In this case, the standard code overestimated migration to the extent that the surface concentrations calculated were virtually nil, whereas they remained high in reality.
Both to account for the respective influence of each parameter and to obtain values that correspond to the natural conditions encountered, the CATHY model was reused with non-standard parameters, in particular a lower K_d for NRV 26 to take the presence of peat into account, and a higher strontium K_d for NRV 31. In this last case, two values for the diffusion coefficient of seepage water in the soil were also tested: the standard values and a value fifty times higher.The choice of adjusted parameters can improve the agreement between observation and calculation, but only within certain limits. For example combining increased values for K_d and D (fifth curve, figure 4) is a reasonable compromise, even though it does not succeed in explaining deep migration. Some of the basic hypotheses will undoubtedly have to be revised to produce a new code that, although perhaps more accurate, will also be more complex.

Modeling the soil→plant transfer factor is normally a simple matter. The difficulty often lies in defining the concentration of soil to be used, particularly where radionuclide distribution in the soil is non-uniform. A predictive model using vertical distribution of cesium in soil and a notion of K_d would undoubtedly be more efficient than the over-simple rule of three. The CATHY model could include a module for calculating soil→ plant transfer based on the distribution of radionuclide concentrations in the soil and in rootlets absorbing the soil solution. We need to find ways of expressing soil→soil solution→rootlet transfer that are simple and general enough to be applied.

Conclusions

Analysis of results gives rise to the following general reflections :

- the *in situ* γ spectrometry measurements performed by the IBRAE are valid, even though they appear somewhat to underestimate deposition where radionuclide distribution is non-uniform,
- the majority of radionuclides deposited remain near the surface with a regular decrease in concentrations to a depth of approximately 15 cm,
- cesium migration in the soil of former garden NRV 26, which contained organic matter, is greater than in the podzolic soil of NRV 31, at least to a depth of 14 cm,
- in the podzolic soil of NRV 31, strontium behaved similarly to cesium, for which 95% of the total quantity present was in the first 5 centimeters, strontium being nevertheless present to a depth of 20 cm,
- in this case, the rehabilitation techniques used for cesium can be applied for strontium,
- soil→plant transfer factors are close to those described in the literature [1], except for grasses at sampling point NRV 26, where they were 100 times greater,
- this last difference can partly be explained by higher soil concentrations of cesium in the roots and a lower K_d than in the podzolic soil of NRV 31,
- the CATHY 2.0 predictive model gives predictions that agree well with observations for the cesium in the first few centimeters, but not for strontium, for which the standard parameters must be re-evaluated,
- the diffusion coefficient of seepage water in the soil seems to be an important parameter. The standard value of 1.10^{-7} m^2/s appears overestimated since a value 50 times greater noticeably improves the agreement between observation and calculation for strontium at point NRV 31,
- the value of the apparent K_d is of the order of 500 to 600 for strontium and cesium at point NRV 31 and 200 for cesium in former garden NRV 26.

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RADIATION EXPOSURE OF THE POPULATION IN RUSSIA AFTER THE REACTOR ACCIDENT AT CHERNOBYL - EVALUATION OF THE 5-YEAR DOSE EQUIVALENT COMMITMENTS IN THE DISTRICTS OF KALUGA AND BRYANSK

E. Finke (1), P. Hill (2), R. Hille (2) and P.F. Sauermann (2)

(1) Federal Board of Health, Berlin Department
(2) Research Centre Jülich, Dept. for Safety and Radiation Protection

Summary

The 5-year dose equivalent commitments for the first two 5-year periods after the reactor accident at Chernobyl were estimated for two areas of study with different soil contamination. In the first five years, internal doses several times higher than the external doses were recorded for rural areas with a high proportion of home-grown produce in the diet.

In the first five years the cumulative dose was smaller than 100 mSv at all locations and only exceeded the average limit of 20 mSv per year recommended by the ICRP 60 for persons professionally exposed to radiation in two locations in 1987. However, in both cases the values were below the limit of 50 mSv per year recommended for any single year. In the second five years, the cumulative dose was found to be below the limit of 20 mSv per year at all localities.

1. Introduction

In summer 1991, the Research Centre Jülich carried out a measuring programme in Russia on behalf of the Environment Minister of the Federal Republic of Germany in order to determine the radiation exposure of the population and environment. Soil contamination by Cs-134 and Cs-137 as well as the body burden of the population in the districts of Kaluga and Bryansk were measured as part of the measuring programme /1/.

Applying simple models for the time course of soil contamination and body burden, the 5-year dose equivalent commitments from external and internal radiation were estimated for the first two five-year periods after the reactor accident and assessed on the basis of the limits recommended in ICRP 60.

2. Estimation of Internal Radiation Exposure

2.1 1991 Annual Dose

The annual dose H_1 can be calculated from an individual measurement of the body burden A if it is assumed that the body burden is constant on an annual average. The annual internal dose $H_1(1991)$ for the year 1991 is then given according to:

$$H_{1\ (1991)} = A \cdot (DF) \qquad (1)$$

with the ingestion dose factor (DF) of $4.0 \cdot 10^{-2} \mu Sv/Bq \cdot a$ for adults and the mixture of Cs-134 and Cs-137 on 1.7.1991 /2/.

Using body burden data gathered as part of the measuring programme undertaken by the Federal Republic of Germany, the annual internal doses H_1 (1991) were determined from the average ingestion dose for ten localities each in the districts of Kaluga and Bryansk /2/.

2.2. 5-Year Dose Equivalent Commitments

The measuring programme of the Federal Republic of Germany was undertaken in an area of study lying between the boundary to Byelorussia and Tula in the Russian Federation. The radioactive soil contamination from Cs-137 in this area was between 1 and 30 Ci per km^2, i.e. between $3.7 \cdot 10^4$ and $1.1 \cdot 10^6$ Bq per m^2. It was thus in the same order of magnitude as the contamination values measured in the region of Berchtesgaden in 1986 /2/. It would therefore seem appropriate to base an estimation of the ingestion dose of the Russian population in the first two five-year periods after the reactor accident

in Chernobyl on the time curve of the ingestion dose received by the population in Berchtesgaden in the years 1986 - 1990.

By analogy to an estimation of the full lifetime equivalent dose in /2/, the five-year dose $H_5(R)$ can be calculated

according to $$H_5(R) = H_5(B) \cdot \frac{H_1(R)}{H_1(B)} \quad (2)$$

with $$H_1(R) = A \cdot (DF)_a \quad (1)$$

resulting in the following

$$H_5(R) = \frac{H_5(B) \cdot (DF)_a}{H_1(B)} \cdot A$$

$$= C \cdot A \quad (3)$$

A is here the average ambient dose of the measured body burdens. For adults, C results as the average of the C values calculated for men and women /3/ as

$C_1 = 1.88$ (1st five-year period)
and $C_2 = 0.1$ (2nd five-year period).

3. Estimation of External Radiation Exposure

3.1. 1991 Annual Dose

The determination of the 1991 annual external dose is based on measurements of soil contamination with Cs-137 /4/.

The method used by us for dose reconstruction is based, according to /5/, on the results of more than 10,000 direct measurements of the external radiation dose using thermoluminescence dosemeters implemented in areas with surface contaminations of 1 - 60 Ci/km² in the years 1989 and 1990.

The annual external dose $H_e(1991)$ is accordingly determined by the following simple relation:

$$H_e(1991) = \frac{A}{F} \cdot (DF)_F \quad (4)$$

where $H_e(1991)$ is given in rem/a,

A in curie
F in km² and the
surface dose factor $(DF)_F$ in

$$\frac{\text{rem} \cdot \text{km}^2}{\text{a} \cdot \text{Ci}}$$

The values of $(DF)_F$ prescribed according to /5/ amount to the following for 1991:

a.) 0.013	$\frac{\text{rem} \cdot \text{km}^2}{\text{a} \cdot \text{Ci}}$	for villages and localities outside those towns specified in b.) and c.)
b.) 0.009	"	for towns and administrative centres
c.) 0.006	"	for district capitals.

Using the values measured within the framework of the measuring programme of the Federal Republic of Germany and given in /4/ for soil contamination with Cs-137, the annual external doses H_e (1991) for ten locations each in the districts of Kaluga and Bryansk /3/ were determined by equation (4).

3.2 5-Year Dose Equivalent Commitments

Since the element caesium, and thus also the isotopes Cs-134 and Cs-137, is strongly absorbed on clay minerals in the soil /6/, it is still largely present in the uppermost soil layers. In the case of undisturbed rural soils no reduction in soil contamination going beyond the natural decay rate of about 8 % per year is to be expected.

However, in town environments and with the application of soil decontamination methods annual decreases in soil contamination of up to 30 % per year are possible /7/.

In our estimation of external radiation exposure in the two rural areas of Kaluga and Bryansk we assume a decrease of 10 % per year and thus remain on the safe side.

Starting from the measured values for soil contamination in 1991 and the annual external doses calculated from them, the time course of the annual external dose was estimated using

$$H_n = H_o \cdot q^n \qquad (5)$$

$$q = 1 + \frac{p}{100}$$

where n is the year and p = -10 % is the decrease of the external dose per year, and the five-year equivalent dose commitments for the first two five-year periods were calculated.

4. Results

The results are summarized in Tables 1 and 2.

It can clearly be seen that the residents of localities in the Kaluga district received or will receive considerably lower five-year doses than the residents of localities in the Bryansk district.

It is further striking that for most locations the internal dose was greater than the external dose in the first five-year period, whereas the external dose will be greater for all localities during the second five-year period.

If the cumulative doses are assessed on the basis of regulations given by the ICRP in its publication 60 /8/, see Table 3, then it is apparent that in all locations studied the population received cumulative doses in the first five-year period below the limit of 100 mSv recommended for persons professionally exposed to radiation. However, in the case of Unetscha and Weprin, the recommended annual dose of 20 mSv was clearly exceeded in 1987 with values of 37 and 28 mSv respectively. Nevertheless, even these values are far below the limit of 50 mSv per year recommended for single years.

In ICRP 60, a limit of 1 mSv per year, i.e. 5 mSv per five-year period, is recommended for the general public. This value will only be slightly exceeded in one locality in the Kaluga district in the second five-year period. In contrast, in the Bryansk region there are a number of localities where this value is clearly exceeded. It is recommended that the population living here should be monitored medically and dosimetrically in the same way as persons professionally exposed to radiation.

Recommendations for setting up the required medical facilities are given in /9/.

Table 1: Total dose commitment to adults in the first two five-year periods after the reactor accident in Chernobyl in some places of the district of Kaluga

		1. five-year dose mSv			2. five-year dose mSv		
place	population	external	internal	sum	external	internal	sum
Chwastowitschi	5500	0.5	2.7	3.2	0.3	0.2	0.5
Dudorowski	1200	3.0	6.7	9.7	1.9	0.4	2.3
Jelenskij	2300	7.8	8.5	16.3	4.9	0.5	5.4
Kolodjassi	500	7.2	4.2	11.4	4.6	0.2	4.8
Krasnoje	2000	0.2	4.5	4.7	0.1	0.3	0.4
Kzyn	200	3.8	2.8	6.6	2.5	0.2	2.7
Ljudinovo	45000	1.1	1.6	2.7	0.7	0.1	0.8
Schisdra	5400	0.8	1.6	2.4	0.5	0.1	0.6
Uljanovo	2000	3.3	1.7	5.0	2.1	0.1	2.2
Wotkino	325	2.1	3.1	5.2	1.3	0.2	1.5

Table 2: Total dose commitment to adults in the first two five-year periods after the reactor accident in Chernobyl in some places of the district of Brjansk

		1. five-year dose mSv			2. five-year dose mSv		
place	population	external	internal	sum	external	internal	sum
Guta-Koretzkaja	600	21.6	24.6	46.2	13.9	1.3	15.2
Kivai	1650	8.4	5.5	13.9	5.4	0.3	5.7
Klincy	80000	3.0	2.1	5.1	1.9	0.1	2.0
Lopatni	1400	7.4	18.2	25.6	4.8	1.0	5.8
Malaja Topal	780	5.4	3.7	9.1	3.5	0.2	3.7
Pestschanka	480	1.6	9.3	10.9	1.0	0.5	1.5
Roschnij	1200	12.2	24.6	36.8	7.8	1.3	9.1
Unetscha	260	11.1	78.1	89.2	7.1	4.2	11.3
Uschtscherpje	1600	12.7	34.1	46.8	8.2	1.8	10.0
Weprin	250	23.8	46.4	70.2	15.3	2.5	17.8

Table 3: Recommended dose limits [8]

	dose limit	
application	occupational	public
effective dose	20 mSv per year averaged over 5 years ‹ 50 mSv in any single year	1 mSv

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ORGANIZATION OF A REHABILITATION WORK CENTER FOR A CONTAMINATED SITE OVER A LARGE AREA

M. Jacques PENNEROUX
Atomic Energy Commission (France).

Abstract :

A rehabilitation work centre for a contaminated site over a large aera, and the complexity of such a centre, will require the setting up of a very important organization, strict and inevitably thought out ahead, because it is likely to last for several months, if not several years. It must take into account, in particular :
- The installation of several hundred workers, and an important depot of decontamination equipment and civil engineering devices ;
- The stock of waste produced and its evacuation (the volume can be important).

A great number of French and foreign documentation treats decontamination techniques for radioactivity in the environnement, and the organization of a response team at the reaction phase.
Very few papers treat the organization of a rehabilitation centre.
This is the object of this document.

1. Reminders :

Further to an accident of a radiological character, the following three phases can be noted :
- the reflex phase (or immediate reaction phase) ;
- the reflective phase (or analysis phase of the situation) ;
- the rehabilitation phase.

1.1 The reflex phase :

Corresponds to :

- the setting up of first assistance (fire, medical ...) ;
- the transmission of the alarm to civil authorities (and eventually to military authorities) ;
- the marking out of the contaminated area ;
- the assessment of the fallout charts ;
- first release to the press ;
- starting evacuation of people from the contaminated zone.

1.2 The reflective phase :

This phase corresponds to the analysis and stabilization of the situation.

a) The analysis corresponds essentially to a diagnostic, that is to say :

- specifying the source term ;
- assessing the radiological risk by analysing the results of measurements already carried out on the field and in the environment, and by studying risks of contamination which has been deposited on the ground being transferred elsewhere ;
- determining the trajectory of the cloud and its associated fallouts. This determination can be carried out with the help of calculation codes, and can also be made during the reflex phase. Delimitation is specified by an air mapping system, and by in situ measurements ;
- gathering together specific information of the damaged site : types of dwellings, road network, farming, agriculture, vegetation, hydrographical networks, demography, medical and hospital infrastructure, etc ...;
- defining or confirming the action levels which are to be considered as being indicative intervention levels allowing for the determination of counter measures as a whole which are necessary as first actions for rehabilitation ;
- setting up permanent radiological surveyance of the site and environment.

b) The stabilizing of the situation corresponds essentially to the following actions :

- ensuring the functioning of all security posts : fire, guarding, transmissions, etc ... ;
- organizing command posts, as well as entrance and exit of personnel and equipment for intervention in the contaminated zone ;
- finishing evacuation operations of persons in the contaminated zone and ensure their medical and radiological follow up ;
- finishing marking and guarding of the contaminated zone ;
- limiting the radiological consequences of the accident by temporary containment of contamination deposited on the ground and other surfaces ;
- determining decontamination techniques which are to be set up ;
- starting to give out information to the public, which must be done as soon as possible.

2. Organization of the rehabilitation work site.

The reflex phase and the reflective phase having been carried out, it is then possible to start the rehabilitation phase itself, wich corresponds to counter measures to be taken to restore all, or part of, the area affected by radioactive deposits (agricultural ground, dwellings, various buildings, ...) which has no radiological risk to human beings, that is to say, to bring back contamination levels to pre-established values (or values guides). These criteria can eventually be adjusted by competent authorities (in France the SCPRI (the Ionising Health Physics Centre) advised on by the National Committee of Medical Experts).
The rehabilitation work site will consist of (annex 1) :
- an outside command post ;
- an advanced zone situated at the frontier of the contaminated zone ;
- a no man's land or security zone ;
- the contaminated zone which has to be cleaned up.

To describe this in more detail :

a) The outside command post :

Globally manages all rehabilitation operations for the damaged site.
Has missions to :
- control the rehabilitation operations, in liaison with national authorities and the advanced command post ;
- organizes the release of information.

For this, there are severals units :
- a command unit which advises the person responsible for all rehabilitation operations on the site and who controls organization of the outside command post ;
- a transmission unit, which ensures the liaison between the local and different national authorities (civil and military) and with the advanced command post. It controls the different transmission means and frequences ;
- a unit for anticipated operations, which establishes the work projects, bearing in mind fixed priorities, the weather forecast, the type of land, the work already carried out, the evolution of the radiological situation. This unit is composed of : a "weather forecast" office, in liaison with the National Weather Forecast Office ; a "land-study" office, which is in charge of hydrological studies of the land (points where water collects, nature of land ...) further to information from maps given by specialized services ; an "evaluation of results and work prevision" office , which establishes the rehabilitation work plans further to reports given by work sites and research done by other different offices ; a "waste management" office, which controls all problems linked to packaging of waste, and provisional or definitive storage of these wastes ; a "radiological situation" office, which keeps up to date the radiological situation of the site and its environment further to measurements carried out on the contaminated zone and outside the contaminated zone.
- a unit which controls the personnel, equipment and provisioning of water and energy to the work site. This unit is composed of : a "budget accountability" office which controls the budget and ensures the accountability of the site as a whole ; a "personnel" office, which is in charge of organizing the recruiting of specialized personnel, according to the needs of the site, and which centralizes the follow-up of personnel on the medical side, on the radiological side (results of analyses, tests ...), and on the professional side (time spent working in contaminated areas, rest periods, holidays ...) ; an

"equipment" office, which ensures the purchasing and follow-up of material necessary for the good function of the work site, and in liaison with the "budget accountability" office ; an "energy-water" office, which controls the supply of energy (hydrocarbures and electircity) and water to the site ; a "free zone" which controls the supply of food, of organizing lodging for the personnel (in liaison with civil and military authorities), and organizing transportation for personnel.

- a "public relations information" unit ensures the release of information to public authorities, local candidates and the population concerned, the general public and the media. This mission, of capital importance in the management of a crisis caused by a radiological accident, must be considered as a permanent concern for the responsible authorities. This unit is composed of : information for public authorities" office, whose work it is to write out technical and operational reports to be sent to ministerial authorities, both civil and military. It also has to organize visits for important persons to the site ; a "public relations" office, which centralizes information concerning the site, in liaison with the other units, draws up press communications and organizes their release, prepares periodical press conferences, ensures information is given to personnel on site ; a "coordination of information" office which verifies the compatibility of information from the different organizations and controls the correct release and interpretation of the communiqués.

b) The advanced zone (annex 2).

This is an area near the contaminated zone which must be placed a-weather to dominant winds.
The necessary means as a whole for the functioning of the different sites are present in this zone, which includes :

- An advanced command post, which ensures liaisons with the outside command post and which coordinates the functioning, security and activity of the different sites. It is composed of a "transmissions" office, which ensures the permanence of liaisons with the outside command post and with the site as a whole (release of alarms, communications linked to the work : work and security frequences). It is indispensible to insist on the importance of quality given to transmissions between the different intervening parties on the site : this is a priority. A "work organization" office which organizes, from decisions given by the outside command post, the distribution of tasks between the different teams. It establishes daily a report on work carried out. An "equipment" office which keeps up to date the availability of the personnel and equipment. It centralizes the requests for material necessary ; a "waste" office which controls all problems related to the treatment and storage of waste produced (in accordance with decisions taken at the outside command post) ; an "outside security" office, which ensures the technical surveyance of the site by a video network or by obeservers.
- A control and security post which ensures the protection of the contaminated zone and free zone against any penetration of non-entitled persons (24h/24h).
- A group for storage, maintenance and preparation of materials, which is composed of : a team in charge of storage and distribution of materials, fuel, water ...) ; a team for preparing equipment, maintenance and repairs (personnel specialized in mechanics, electronics, transmissions) ; a "research" team to develop, in liaison with the staff working on the site, all equipment adapted to the speciality of certain work.
- A storage area for packaged waste.
- A teaching site to train the staff for work in the contaminated area, particularly for certain work which necessitates preparation or training.
- A radiological control unit, whose work it is to carry out and analyse tests made on personnel, equipment and the environment, to follow up in real time the dosimetry of the intervention team. This unit is composed of : a radioactivity analysis laboratory which treats only urgent analyses. The other tests are sent to specialized laboratories ; mobile health physics teams to ensure the surveyance and sampling on the field.
- A medical centre, which : ensures the daily follow up of the personnel working in the contaminated zone as well as in the nearby non-contaminated zone ; gives first aid in the event of injuries, and ensures, if necessary, evacuation to the nearest hospital.
- A centre for treating contaminated clothing.
- A free zone, which regroups all the necessary means for meals and for the relaxation time for personnel working on the site.

c) The no man's land zone.

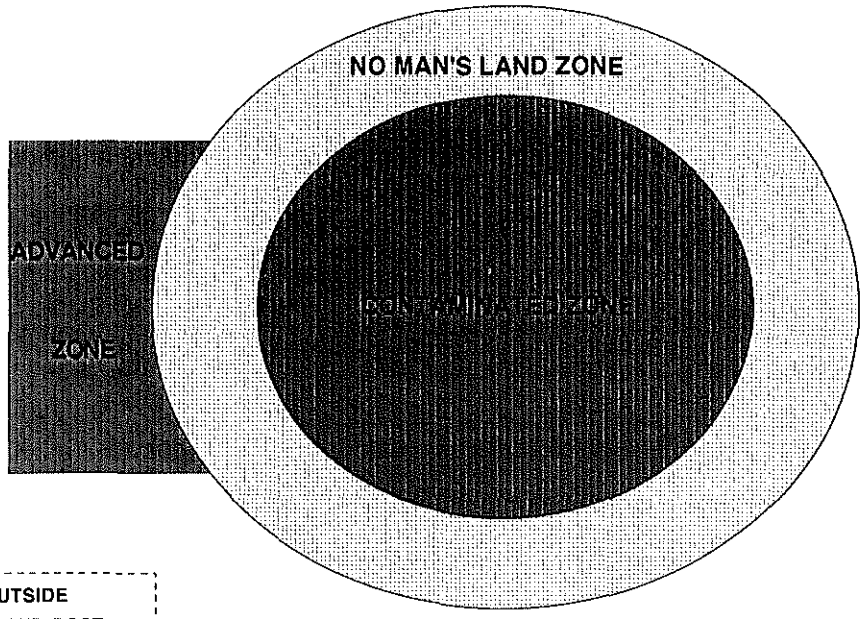
This zone is made up of :

- An intermediary zone for the control and decontamination of staff and equipment before their transfer to the advanced non-contaminated zone, and the contaminated zone.

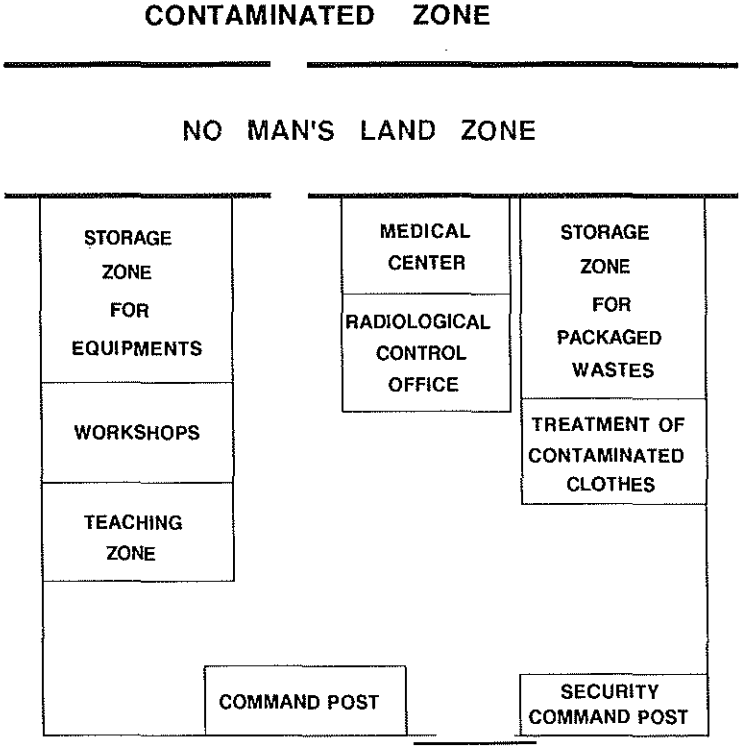
- A security zone around the contaminated zone in the event of contamination being extended. Its size depends on weather conditions at the time, and nature of the ground . This zone includes : an area for personnel control, an area for equipment control, and an area for waste treatment (annex 3).

d) The contaminated zone (annex 4).

The contaminated zone is divided up into a certain number of work sites or areas of activity for which the nature of the work will depend on : contamination levels, nature of the ground, relief and imposed priorities.
Each of the work sites will be fitted out for each particular job : a temporary centre for waste produced, an improvised decontamination centre situated near the sites, and a working area.
An area for temporary storage of all waste produced, while awaiting their packaging, and an area for storing and repairing contaminated equipment will be fitted out near the contaminated site.
It is important to note that the decontamination techniques set up must imperatively consider the problem of waste produced and so be thoroughly investigated so as to limit to a maximum the quantities of waste produced. The waste comes most particularly from the decontamination of the ground, but also from the work area : used clothing, effluents, equipment, etc ...
The supply of energy and water to the different sites will be made either by a system of cables and pipelines coming from the non-contaminated advanced zone, or by tanks or drums and generating plants.
The access to the work sites will be made by vehicles or by foot, and following a marked route. The vehicles will be parked, at the end of the day, in a special car park situated nearby the contaminated zone.

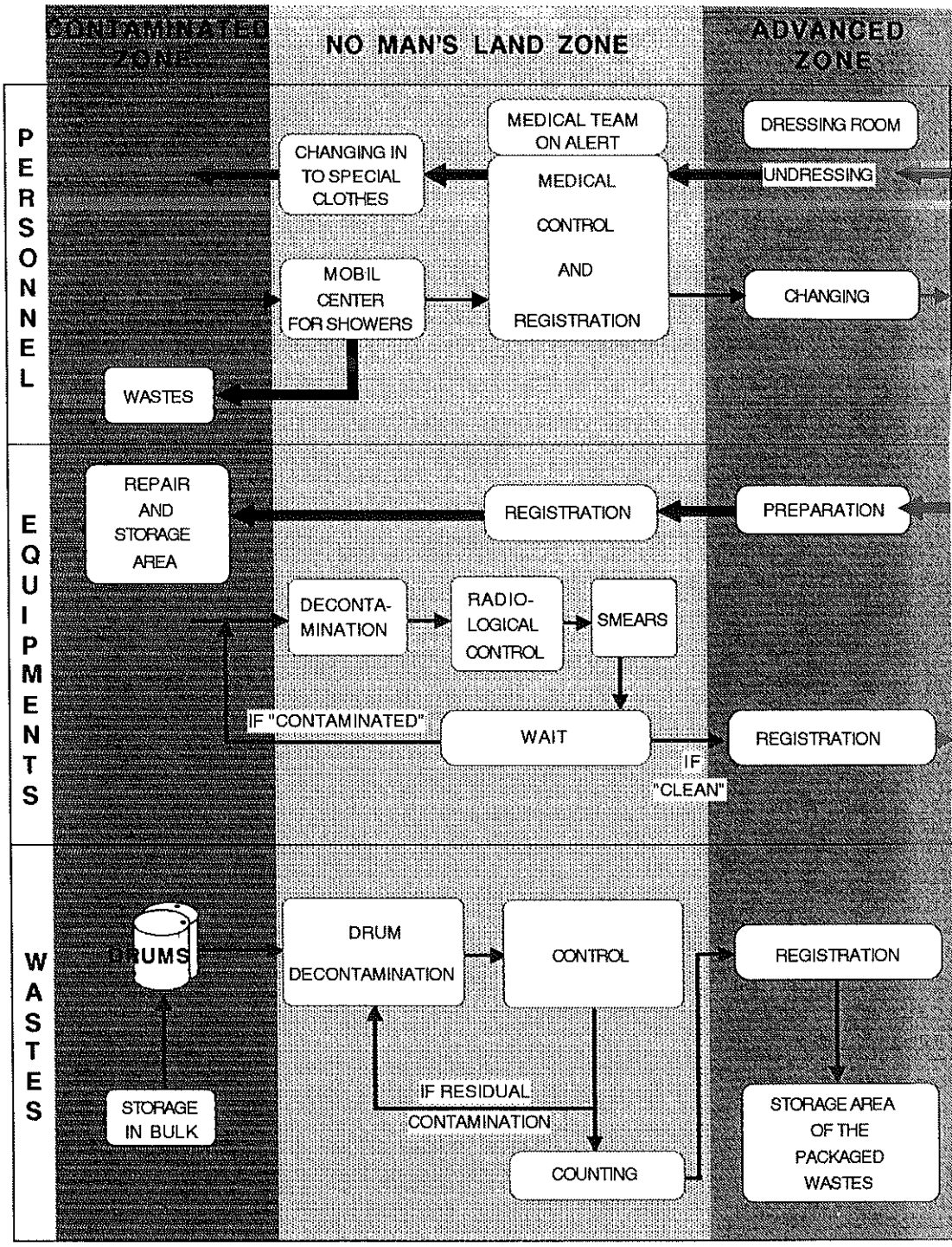


ANNEXE 1



ANNEXE 2

ANNEXE 3



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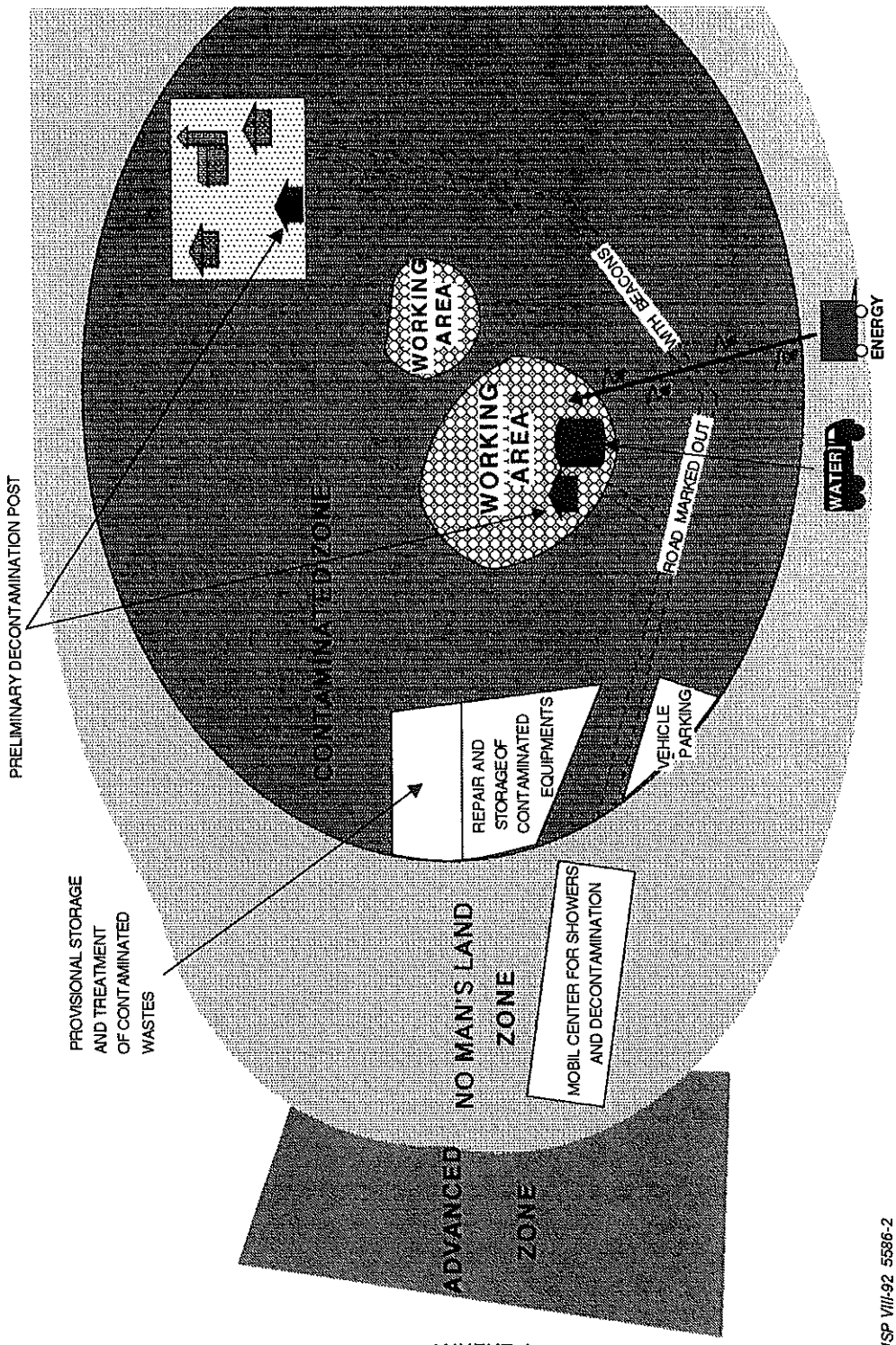

ANNEXE 4

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WASTE MANAGEMENT OF CONTAMINATED AGRICULTURAL PRODUCTS AFTER RELEASE OF RADIOACTIVE CESIUM FROM NUCLEAR PLANTS

E. Bayer*, T. Haug, M. Kutubuddin* and H.-J. Reinecke
Department of Organic Chemistry* and Isotope Laboratory,
University of Tübingen, Germany

Summary

A procedure is described by which it is possible to dissolve and concentrate radioactive cesium from contaminated agricultural products after thermocatalytic conversion and, therefore, substantially reduce the amount of radioactive waste.
The thermocatalytic low temperature conversion in the range of 280°C to 380°C converts biomass (in this case whey powder) to oil, fatty acids and coal. After the conversion the initial ^{134}Cs and ^{137}Cs activity is entirely found in the coal. Cesium is extracted with water or aqueous solvents from the coal and is subsequently separated from the liquid with potassium iron (III) hexacyano ferrate (II) (Prussian blue). This procedure allows a decontamination of the whey powder up to 98 %. The resulting radioactive waste contains no fermentable organic compounds, so that a subsequent waste treatment causes no difficulties. The method described here is effective, economical and requires minimal apparative design. Moreover the other resulting products can easily be disposed, or can be used for other purposes.

1. Introduction

After the Chernobyl accident in May 1986, there existed a considerable amount of agricultural products contaminated with radionuclides in Germany [1]. Contaminated milk was mainly found in southern Germany. Since by milk processing the main part of ^{134}Cs and ^{137}Cs remains in the whey, contaminated milk was preferably worked up to milk products like butter and cheese. 5000 t of contaminated whey powder (initial specific activity of cesium about 5000 Bq/kg) posed a problem for the waste management in the Federal Republic of Germany in 1987.

Using a special method, the whey powder was decontaminated in the following years [2]. For decontamination the whey powder was dissolved in water and the radioactive cesium was separated using the technique of ultrafiltration. However, using the ultrafiltration method a complex process was required to prevent the building of secondary membranes in the whey concentrate.

Therefore we applied another procedure for the decontamination of whey which we present here. The decontamination is carried out by the low temperature conversion of the whey powder to coal followed by an extraction with water or aqueous solvents and the selective adsorption of cesium on potassium iron (III) hexacyano ferrate (II) [3-6].

2. Experimental

The process of decontamination consists of three steps as shown in *Figure 1*.
100 to 500 g of the contaminated whey powder was converted thermocatalytically in an oven for discontinuous low temperature

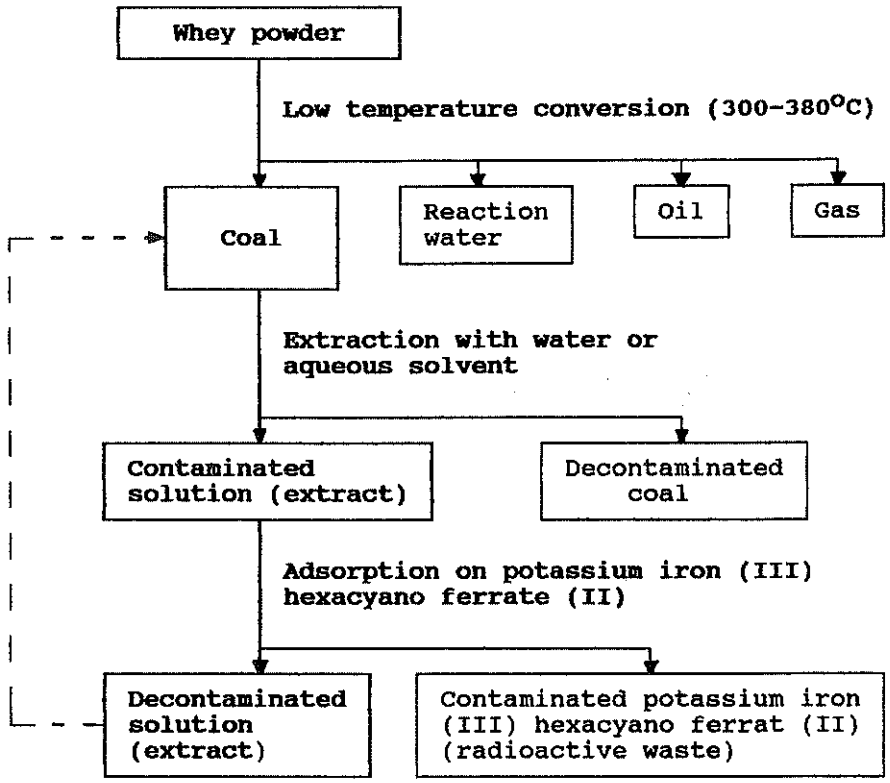

Figure 1: Procedure for decontamination of biological material (whey)

conversion at 300-380°C [3]. The proteins and lipids are converted into oils, fatty acids, coal, water, and a mixture of gases containing ammonia and carbon dioxide. Carbohydrates are converted into coal which is the main product.
The coal is then ground and stirred in water for several hours and filtrated. In some cases various amounts of acids are also added.
The contaminated solution (extract) is subsequently applied directly to a column (1 x 10 cm) filled with 1 g potassium iron (III) hexacyano ferrate (II), which selectively adsorbs cesium. The extract is allowed to run through the column with a flow rate of 40 to 100 ml per hour. The capacity of the potassium iron (III) hexacyano ferrate (II) (purchased from the Heyl company) is ascertained by activity determination using a neutral solution of ^{137}Cs.

The activity of ^{134}Cs and ^{137}Cs containing samples is ascertained in polyethylene bottles with a HPGe semiconductor detector (40% efficiency) from EG&G Ortec company. The efficiency is determined using calibrated standard solutions from the Physikalisch Technische Bundesanstalt.

3. Results

3.1. Low temperature conversion

By the low temperature conversion, the whey powder (about 1.6 kBq/kg ^{137}Cs; $^{137}Cs/^{134}Cs$ (5:1) at the time of investigations) is converted to the following products shown in *Table 1*.

Table 1: Conversion products from 500 g (700 ml) whey powder; conversion temperature, 380°C; conversion time, 3 h; the values are given in %.

Conversion products	Volume	Weight
Reaction water	19	28
Oil	6	7
Coal	46	46
Gas mixture	-	18 *)

*) The value is determined from the difference

After the thermocatalytic conversion of whey powder, the initial cesium activity is totally found in the coal. The ^{137}Cs activity in the other conversion products is below the detection limit of 0.2 Bq of the applied measurement procedure.

Decontamination of the coal increases with the conversion temperature as shown in *Table 2*. To compare the result of decontamination, a decontamination ratio is defined. The decontamination ratio is the ratio of ^{137}Cs activity of the contaminated aqueous solution (extract) and ^{137}Cs activity of the whey powder.

Table 2: Dependence of the decontamination ratio on the conversion time and temperature.
Activity values are based on 100 g whey powder; extracting solvent, water; volume, 150 ml; ratio of water/coal, about 3.5:1; extraction time, 20 h.

^{137}Cs (whey) (Bq)	Conversion temperature (°C)	Conversion time (h)	^{137}Cs (extract) (Bq)	Decontamination ratio
161	300	3	132	0.82
157	350	3	140	0.89
154	380	3	146	0.94
160	380	5	149	0.93

3.2. Extraction

The decontamination ratio depends on the chosen ratio of water/coal. The results are shown in *Table 3*.

Table 3: Dependence of the decontamination ratio on the amount of water.
Activity values are based on 100 g whey powder; extraction time, 20 h; conversion temperature, 350°C; conversion time, 3 h.

^{137}Cs (whey) (Bq)	Water (ml)	Coal (g)	^{137}Cs (extract) (Bq)	Decontamination ratio
165	40	46.5	131	0.79
165	70	46.5	140	0.84
165	100	46.5	142	0.86
157	150	45	140	0.89
157	200	45	141	0.90
157	300	45	143	0.91

Extraction time has also an influence on the decontamination. More than 80 % of the ^{137}Cs activity is extracted after a short time. If the coal is stirred in water for 5 h up to 3 days, the decontamination ratio will be increased from 0.84 to 0.96.

Additionally, the influence of different concentrations of hydrochloric acid in the extracting solvent on the decontamination is investigated. The results are shown in *Table 4*.

Table 4: Dependence of the decontamination ratio on the HCl concentration of the extracting solvent.
Activity values are based on 100 g whey powder; extraction time, 20 h; solvent/coal ratio, 1.5:1; conversion temperature, 380°C; conversion time, 3 h.

^{137}Cs (whey) (Bq)	HCl concentration of the extracting solvent	pH-Value (extract) (pH)	^{137}Cs (extract) (Bq)	Decontamination ratio
164	H_2O	10.3	138	0.85
164	0.001 N	10.0	140	0.85
164	0.01 N	9.8	140	0.85
164	0.1 N	9.5	139	0.85
164	1.0 N	5.3	156	0.95

By repeating the extraction procedure using 1 M hydrochloric acid, the decontamination ratio can be increased to 0.98.

The influence of the complex forming organic acids on the decontamination is investigated in other experiments. Using an aqueous extracting solvent containing 10% organic acid (acetic acid, tatratic acid, or citric acid), a decontamination ratio of > 0.90 is resulted. Extraction using hydrochloric acid results in the highest decontamination ratio of 0.97 as shown in *Table 5*.

Table 5: Dependence of the decontamination ratio on different acids.
Activity values are based on 100 g whey powder; extraction time, 20 h; solvent/coal ratio, 1.5:1; 10% acid; conversion temperature, 380°C; conversion time, 3h.

^{137}Cs (whey) (Bq)	Acids used in the extracting solvent	pH-Values extracting solvent	pH-Values extract	^{137}Cs (extract) (Bq)	Decontamination ratio
150	acetic acid	1.6	4.5	139	0.92
150	tartaric acid	1	3.5	137	0.91
150	citric acid	1.2	4.7	139	0.93
150	hydrochloric acid	< 0	< 0	146	0.97

3.4. Removal of the radiocesium

More than 98 % of the ^{137}Cs activity of the contaminated extract of 2000 g converted whey powder is retained on the column filled with 1 g potassium iron (III) hexacyano ferrate (II). The cesium capacity amounts to about 100 µmol cesium per g of potassium iron (III) hexacyano ferrate (II).
Because of the additional extraction of organic and anorganic salts of the coal, the capacity of 1 g potassium iron (III) hexacyano ferrate (II) is diminished after eluting contaminated extracts of 2300 g of the converted whey powder.

4. Discussion

2 kg of whey powder can be decontaminated with 1 g potassium iron (III) hexacyano ferrate (II).
The specific ^{137}Cs activity of potassium iron (III) hexacyano ferrate (II) (decontamination ratio 0.94) amounts to 3 kBq/g, and the decontaminated coal to 0.1 Bq/g. - The value for the release of the coal amounts to 50 Bq/g ^{137}Cs according to 10^{-4}-fold of the exemption limit of "Anlage IV Tabelle IV 1, Strahlenschutzverordnung (BRD)", and 6,2 Bq/g for weak radioactive waste according to Wirth [7]. Therefore, the coal can be recycled without problems. - Using this decontamination process a weight reduction of the radioactive waste of a factor 2000 is achievable.

The great advantage of decontamination via low temperature conversion is the simple extraction of cesium from the coal and its adsorption on potassium iron (III) hexacyano ferrate (II). Since the low temperature conversion converts organic compounds of the whey powder such as proteins into coal, oil, water and a mixture of gases, the coal now contains no organic compounds which might disturb the decontamination procedure. Because of the volatility of the cesium, combustion or pyrolysis at higher temperatures cannot be considered for destructing organic substances. A substantial advantage of the conversion method over pyrolysis is that the formation of cancerogene aromatics, toxic acetonitrile, tar, dioxins and volatile metals is avoided using the conversion pro-

cess. Therefore, low temperature conversion, in the range up to maximum 380°C, is the best method to release cesium from the organic substances.
Beyond that, little water is needed since the aqueous extracts can be recycled after the separation of the radioactive cesium on the column (*Figure 1*). However, after several extractions, the increasing amount of salts in the extract leads to reduction of the capacity of potassium iron (III) hexacyano ferrate (II). These salts can be removed from the recycling process using an additional mixed bed ion exchanger. An additional advantage is that the low temperature conversion can be applied to other agricultural products or waste and sewage sludge as well [4].

In contrast to the decontamination by ultrafiltration, this decontamination process can be carried out in small facilities at low cost. In addition, the side products of the low temperature conversion such as coal and oil can be used for energy production.

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AGRICULTURAL SOILS DECONTAMINATION TECHNIQUES; METHODS AND RESULTS OF TESTS REALISED NEAR CHERNOBYL

Henri MAUBERT, André JOUVE, Naoll MARY
Institut de Protection et de Sûreté Nucléaire, Département de Protection de l'Environnement et des Installations., IPSN/CEA, 13108 St Paul lez Durance, France

Rocio MILLAN-GOMEZ
Centro de Investigaciones Energeticas, Medioambientales y Tecnologicas, 22, Avenue Complutense, 28040, Madrid, Espana

Summary

After a major nuclear accident, decontamination of agricultural soils would be necessary in order to reclaim the land. Specific techniques were studied in the framework of the European program for Rehabilitation of Soils and Surfaces after an Accident (RESSAC). Different ways to remove the top layer of soils are described, and especially the use of Decontaminating Vegetal Network (D.V.N.) combined with spraying of organic polymers.

Real scale tests in the 30 km zone around the Chernobyl nuclear power plant showed that it is possible to achieve an excellent decontamination of agricultural fields (decontamination factor greater than 95%).

1. Introduction

Although every precaution has been taken in France and more generally in the Western world, during the design, construction and operation of nuclear reactors, the possible occurrence of serious accidents resulting in a considerable discharge of radioactive products into atmosphere cannot be completely ruled out. It has therefore been considered necessary to draw up emergency plans to protect the population against the effects of such accidents.

As regards the actual accident phase and the period immediately following it, these plans are already operational. The French authorities have also studied a Post-accident Action Plan to be applied over a longer period following the accident. Its purpose is to facilitate a return to normal living conditions in the areas affected.

The program for Rehabilitation of soils and Surfaces after an Accident (RESSAC) is aimed at collecting technical data related to the transfer of radionuclides and the methods required to reach the above-defined objectives. It is supported at the national level by the Institute of Nuclear Protection and Safety (IPSN) which provides the main source of finances. At international level it is one of main Radioprotection programs of the Commission of the European Communities. Many countries are actively involved in this research including Belgium, Italy, Spain, Great-Britain and Germany. The program started in 1985, that is to say before the Chernobyl accident occurred.

In the framework of an agreement between the Commission of the European Communities and the states of Russia, Ukraine and Belarus, a collaboration with organisations from these three Republics is now operational. The operator in the 30 km zone around the Chernobyl power plant is the CHECIR (Chernobyl Centre for International Research). This collaboration program is often referred at as the CEC/CHECIR Agreement. It allowed to test on a real scale some of the methods investigated.

2. Rehabilitation method

In this paper we will only discuss the possibility of soil decontamination, i.e.actual removal of top soil layers by physical methods.The most immediate method for top soil removal is to use public works machinery. However there are undesirable side effects: it is very difficult to remove thin layers that way, it makes contaminated dust, it may reduce the fertility and a tremendous amount of wastes are produced. After a nuclear accident most radionuclides of interest penetrate very slowly into the soils [3]. Techniques allowing to cover the soil and then to remove only a thin layer (1 to 3 cm) were developed They include the use of Polyurethane foams and strippable paints, wich has been described elsewhere [4].

These actions will generate wastes. It is therefore necessary to implement in the field some sort of pretreatment allowing a reduction of mass and volume along with some extraction of the radioactivity under a liquid form easier to concentrate. Microbial digestion is investigated to perform this work.

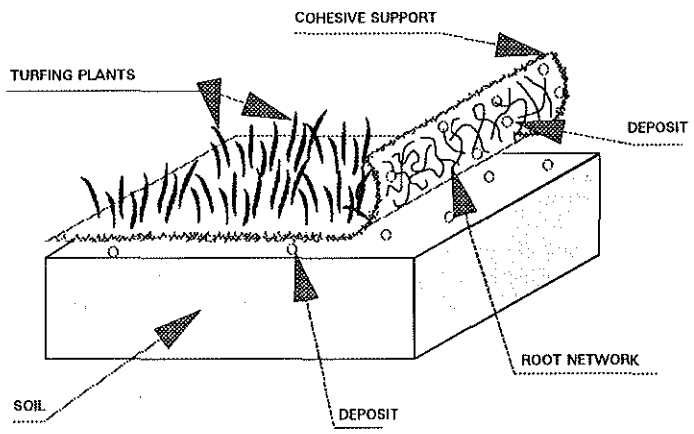
In this paper two original methods will be discussed:

Decontaminating Vegetal Network (D.V.N)

This method uses turfing plants producing a mat layer over the soil which may be removed together with a soil layer, by means of a sod harvester. The principle of the D.V.N is to grow grass onto the contaminated area without tillage of the soil, on a cohesive support which will allow a fast and good growth of the plants with a dense root-network able to entrap the contaminated soil particles [5].

The procedure is derived from existing techniques of hydro-seeding which allows to sow sod seeds together with a growing medium made of a mixture of peat, polysaccharides, fertilisers and water. This technique is used to grow grass to prevent erosion on ski tracks, highway sides... It may facilitate the operational scattering of the D.V.N especially respecting radioprotection criteria. The hydro-seeding mixture may be applied by helicopter at a rate of 0,3 km²/day.

Fig 1 : PRINCIPLE OF THE DECONTAMINATING VEGETAL NETWORK



In the second step, the D.V.N. is removed together with the 1 or 2 first centimetres of the soil invaded by the root-network, which contains the most part of the contamination (fig. 1). Since the resuspension problem is partly solved by the D.V.N, and assuming that the migration rate in the soil profile is a slow process, the D.V.N can remain on the contaminated soil for a quite long period of time (about 1 year) without adverse consequences for the success of the rehabilitation. On the contrary, time favours the grass growth and may be used to organise the harvest of the D.V.N. with a turf cutter. These machines already exist at a moderate cost.

The third step will be the storage or the decontamination of the D.V.N. This technique allows to remove a soil layer of minimal thickness, according to decontamination ratio to reach. Therefore the waste volume will be limited. The main part of the waste is organic and could be treated by bio-degradation.

Gels of organic polymers

The possible use of different soil stabilisers for land reclamation and cleaning, on the basis of manufacturers and application data has been discussed before [6]. These soil stabilisers include polysaccharides which are expected to have electrochemical affinities with Cs and Sr, and polyacrylamides used together with polysaccharides as soil conditioners in modern agriculture.

The first step of our work was the selection of different soil conditioners or algal cultures according to their affinity for caesium and strontium in aqueous solutions. In a second step the selected compounds were tested in different application trials: mixed to the soil, sprayed on bare soil surfaces or on plant leaves. These

tests were concluded by finding an interesting soil conditioner for both preventing the soil particle resuspension and allowing the removal of the first millimetre of the contaminated soil by mechanical brushing without dust. It is called PR 3005 and is commercialised by the SNF Floerger[1] firm. It is a reticulated anionic poly-acrylamide. It may absorb 500 times its weight of water, and, at concentrations below 1 g/kg it is fluid enough to be sprayed by agricultural sprinklers such as those used for pesticides. The concentration of PR 3005 in water is 0,7 %, it is equivalent to 13g/m². The cost of PR 3005 is 20 FF/kg.

Its use for rural soils decontamination is protected by a French Patent.

After spraying on the soil it makes a gel film. This film may be removed by a brushing machine such as those used for street cleaning. If the deposition is recent, good decontamination ratios may be achieved.

But, perhaps, the main advantage of this technique is the drastic reduction of resuspension and wind erosion.

Another advantage is a momentary prevention of radionuclides transport by runoff. That may be of interest to prevent a rapid contamination of water bodies.

3. Tests in Chernobyl

Within the framework of the CEC/CHECIR agreement it was decided to test both techniques of Decontaminating Vegetal Network and Organic Polymers near Tchernobyl, in the 30 km zone. The scientific organisations involved were:

- For EC the Institut de Protection et de Sûreté Nucléaire (France, Cadarache) and the Centro de Investigationes Energeticas Medioambientales y Tecnologicas (Spain, Madrid).

- For CIS, The Department of Geography of the Academy of Sciences of Ukraine, and the Institute of Cell Biology and Genetic Engineering both in Kiev.

A mission was done in May 1992, and a radioactivity concentration profile was determined in Bourakovka. It showed that the bulk of the radioactivity is located in the first five centimetres. It proved the feasibility of top soil layer removal for decontamination.

Another mission was done In June 1992. A turf harvester and all the suitable equipment was shipped to Ukraine. Two locations were selected, Chistogalovka, and Bourakovka, both West from the reactor within a 10 km distance. Chistogalovka is the most contaminated place. The soil is a podzol of sandy texture. The locations were chosen in agricultural fields abandoned since the date of the accident, in May 1986. During this period the fields have mainly been colonised by Agropyrum Repens a Graminacea which root system is perfectly adapted to the removal by a turf harvester.

Experimental method

The surfacic radioactivity of soils was measured before and after the passage of the turf harvester. A surface of 10x10 cm² was cut on a depth of 5 cm. The depth of the blade was set at 5 cm also.

After the experiment the samples were divided in two: one half for the EC labs, one half for CIS laboratories, in order to make a comparison of results. Gamma spectrometry and strontium 90 measurements were to be made. At present we only have gamma spectrometry results for the samples brought back in EC.

Each of the 500 cm^3 samples taken back was divided in 5 sub-samples, each of them was being counted on a Ge gamma detector, with a determination of the radioactivity of caesium 137 and 134.

If A is the surfacic radioactivity before cutting, and B the surfacic radioactivity after cutting, the decontamination efficiency is determined by the formula:

$$\text{Decontamination efficiency} = A / (A+B)$$

In this method of calculation, the radioactivity included in the soil below the first 10 cm from the initial ground surface is neglected. But we can see from the soil radioactivity profile, and from the results in table 2 that this an acceptable approximation.

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[1]SNF Floerger, 41 rue Jean-Huss, 42028 St-Etienne, Cedex 1, France

Results

The first results based on caesium determination showed that the decontamination efficiency was greater than 95%. A more detailed analysis was then undertaken for all the radionuclides detectable by gamma spectrometry. The preliminary results are shown in table 1. A determination of 90Sr is being done but we do not have the results at this time.

Table 1 : Surfacic radioactivity of samples (10x10x5 cm^3) before and after top layer cutting in tw locations; ($Bq/100cm^2$); counting uncertainty < =+-10%

	137Cs	134Cs	60Co	106 Ru+Rh	125Sb	144 Ce+Pr	154Eu	241Am
CHISTOGALOVKA Before cutting	20640	1510	34	2858	341	3168	262	148
CHISTOGALOVKA After cutting	192	15,5	0,49	23	7,2	11	0,98	1,3
BOURAKOVKA Before cutting	3933	290	13,7	336	55	352	30	24
BOURAKOVKA After cutting	116	10	-	15	3,5	11	1,2	-

Table 2 : Decontamination efficiency of turf cutting (%)

CHISTOGALOVKA	99	98	98	99	97	99	99	99
BOURAKOVKA	97	96	-	95	93	96	95	-

From these results we can assess that the method is very efficient. If we take for example an area where the 137Cs contamination is 40 Ci/km^2, which was taken by the Soviets as a relocation level, a decontamination efficiency of 97% would bring the surfacic radioactivity down to 1,2 Ci/km^2 (44400 Bq/m^2). That value is below intervention levels in CIS [7].

A countermeasure such as deep ploughing [8], also investigated in the framework of CEC/CHECIR agreement, could then dilute the radioactivity in 1000 kg of soil by m^2. Therefore the concentration of 137Cs in soils would be around 44 Bq/kg of soil. For barley on sandy podzolic soil; Alexakhin [7] gives a transfer factor of 0,2. Notwithstanding the further use of countermeasures such as adequate fertilisation, the radioactivity of grain would then be around 9 Bq/kg, which is more than 100 times below the EC commercialisation levels.

More results from our colleagues from CIS should confirm these assumptions. The area that could be treated that way exceeds 60000 hectares.

After the top soil removal, there is a risk of resuspension for what is left of the radioactivity. Then the spraying an organic gel solution will eliminate that risk at a very low cost. An experiment was done in order to measure resuspension with and without organic polymers application.

Waste treatment

The amount of wastes generated is about 35 to 40 kg/m^2. It could have been much less if the intervention had been done shortly after the accident, but the Chernobyl disaster occurred 6 years ago. These wastes are made of soil particles and organic matter mixed together.

The technical means for the treatment of this contaminated matter should be as simple as possible, excluding options like chemical treatment, incineration, not adapted to huge quantities. Natural biodegradation of the organic matter contained in the wastes will occur without human intervention, if the residues are stored in an adequate manner.

The rolls of soil and grass were put in a hole and isolated from the surroundings by a plastic liner. This hole was then filled with water and covered. It is expected that microbial degradation will break the organic cells walls and liberate radionuclides. Metabolites such as acids and chelating agents could help the solubilisation of radioactivity. Our Ukrainians colleagues will make a survey of the evolution of the experiment. When appropriate, the liquid phase will be extracted and treated for example on ion exchange

resins to concentrate and immobilise the radioactivity in order to put it to a radioactive waste disposal site. The remaining solid matter may be isolated from the environment by geotechnical barriers, such as those used for uranium tailings.

Possible developments

Although this year's experiment is not fully completed we think the we have an efficient method for the decontamination of agricultural soils around Chernobyl. However our present equipment is to small to undertake large scale rehabilitation. Plans were made to build a large turf cutting machine, fully automated, driven by a tractor and pouring the residues in trucks. This machine would be equipped with adequate radioprotection devices in order to reduce the irradiation of workers.

After turf cutting the growth of a new plant cover could by enhanced by the technique of hydro-seeding. That would also prevent resuspension and runoff.

The wastes could be gathered in provisional treatment sites, and after re-concentration and such reduction of mass and volume that can be achieved at a reasonable cost, immobilised by means of geotechnical barriers.

Within the CEC/CHECIR program, a cost-benefit analysis of these possibilities is being done as well as a feasibility study.

4. Conclusion

After a nuclear accident the possibility of rehabilitating heavily contaminated fields does exist. In Chernobyl, the contaminated areas are so large that it is still a great financial and industrial problem.

In the case of an accident on a nuclear power plant equipped with a containment building and a sand filter the radionuclide release would be much smaller than in Chernobyl. From safety studies, only areas below distances of a few kilometres downwind from the reactor would need heavy rehabilitation such as vegetation clearing and top soil removal; it would then be very possible to use the methods described in this paper. In this eventuality the results of the RESSAC program and the CEC/CHECIR agreement show that it is possible to mitigate efficiently the consequences of a nuclear accident.

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EXPERT CONSULTANCY IN AN EMERGENCY, BY IDENTIFYING THE REQUIREMENTS OF THE PUBLIC AUTHORITIES

Jacques NEY
Institut de Protection et de Sûreté Nucléaire, Département de Protection de l'Environnement et des Installations, Centre Technique de Crise - IPSN/CEA B.P. 6 - F92265 FONTENAY AUX ROSES CEDEX

Summary

This document sets out the main objectives of expert consultancy, indicates the main characteristics of appropriate methods and describes the associated flow of information. It attempts to identify the main scientific and technical areas of competence of the experts, while also emphasising that expert consultancy in an accident situation is a true profession, and implies special training.

1. Introduction

In the nuclear domain, the existence of Centres of Expertise working with public authorities is the result of a need expressed by the public authorities and a demand on the part of the general public. It has been thought necessary to have access to the opinion and advice of experts who are independent of the operator, and even of the public authorities, in order to appraise an accident situation and the actions carried out by the operator, forecast its likely development, and prepare for the decisions which must be made by the public authorities relating to protection of the population and the environment.

It should be noted that the specification of tasks appropriate to a Centre of Expertise does not diminish in any way the regulatory responsibilities of the operator.

2. The requirement of public authorities for expert advice

The specification of requirements on behalf of the public authorities should be clear and complete, in order to determine the range of skills which will be required to provide the most appropriate opinions and advice.

In the accident phase (from the occurrence of an abnormal situation giving rise to a risk of external releases of radioactivity to return of the installation to a state referred to as "safe", that is where there is no further risk of significant release).

- the authority responsible for monitoring the safety of the installations must be capable of verifiing that actions undertaken by the operator are in fact aimed at minimising any existing or potential release of radioactive materials.
- the authority responsible for the protection of people and property must provide protection for the population, and must therefore install counter-measure to provide immediate protection and to reduce the risks to which those services who have to take action on the ground are exposed.

Monitoring of the operator, which may last from several hours to as much as several days, must be accomplished in real time.
The setting-up of counter-measures which require between 6 and 12 hours to install, depending on the site and the season, should be started as quickly as possible, since the risk of significant releases is greatest within a period of 24 hours in most cases.

The quality of the advice given is important, since some of the resulting actions will be irreversible and difficult to alter.

In the post-accident phase (which lasts for as long as normal life has not been re-established in the vicinity of the affected installation), the public authorities must arrive at decisions concerning:

- the counter-measures put in place during the accident phase (additional actions, maintenance or not of containment, etc).
- consumption, marketing of agricultural and food products, water, etc.
- the actions to be undertaken in order to reduce the risk of transferring contamination liable to increase the harmful consequences for health or the environment.

Moreover, they must:

- monitor the progress of the actions undertaken by the operator, and adapt or modify them if necessary,
- provide the general public with the information necessary for the welfare of some and as reassurance for others,
- reduce the health consequences for those working in the affected areas.

The resolution of all these problems becomes even more difficult as the situation is worsened or exaggerated by the media, and as national or international politics complicate the decision-making process.
It is therefore vital that technical advice should come from agreed experts who are as independent as possible.

3. The main aims of expert consultancy

The requirements of the public authorities stem from the various objectives, and these must be established by the experts before they are in a position to formulate opinions and advice.

The accident phase

In this phase, the objectives are:

- rapid assessment of the severity of the situation,
- a diagnosis of the state of the installation, together with an examination of the actions undertaken by the operator,
- a forecast of how the situation is likely develop, and an estimate of the associated release,
- calculation of the radiological consequences associated with this release, taking account of the weather forecasts,
- proposals for counter-measures to give immediate protection, and for the preparation of resources aimed at optimising rehabilitation activity.

The post-accident phase

In the course of this phase, the aims must veer toward:

- a rapid initial assessment based on the real-time monitoring of the consequences of the release, associated with surveillance of sensitive points,
- proposal of remedial counter-measures tailored to deal with the results being obtained during this initial effort,
- satisfaction of local demand for a more detailed overall appraisal of the contaminated area, up to the limits of the zone which has been affected by deposits leading to a concentration in products of radioactivity in excess of the limits regulating product marketing,

proposal of counter-measures to minimise the health and economic consequences resulting from contamination of the environment, with the aim of assisting the work of rehabilitating affected land.

4. Expert consultancy - appropriate methods

Exercises and training have indicated that special methods and tools must be developed in order to attain the objectives of the expert consultancy system, and in order to meet the time constraints imposed by the very short setting-up period.
These methods and tools should take the form of assistance with diagnosis of the state of an installation, help with forecasting the progression of this state, and support in decision-making regarding the forecasts of radiological consequences and assessment of the appropriate post-accident counter-measures.

Specification of the most suitable methods and tools for the crisis situation will enable the areas of knowledge needing to be covered by the expert service to be assessed, and will also allow precise specification of the interfaces (responsibilities, information, communication resources, etc.) to the various participants such as the public authorities, the media, and so on, and will thereby enable the nature of the information which will be passing through these interfaces to be discovered.

The advantages of this approach are particularly notable for any interfaces of a technical nature (operator, weather service, environmental services, etc.). The specificity of the information expected by the experts requires special preparation of these services, based on suitable crisis management and methodology.

5. Expert consultancy and areas of knowledge

It is possible to identify six areas of knowledge which must be covered by expert consultancy in respect of a serious accident in a nuclear power reactor:
- operation of the installation,
- transfer of radioactive pollutants in physical media,
- transfer of radioactive pollutants in living media,
- health consequences of radiological origin,
- radiological measurement methods and equipment,
- technical methods for soil decontamination.

The range of this knowledge is extremely wide, with disciplines ranging from medicine to meteorology, where all experts must have a good knowledge of the nuclear business and of the adaptation of their particular disciplines to this field.
In order to obtain expert advice on the safety of installations, the public authorities have set up organisations with responsibility for applying the regulations. These organisations include experts which form the main source of expert advice in an accident situation. However, in addition to the specialisation which their qualifications imply, special crisis training is also needed and continuous training is essential.

It should be noted that training on specific methods and tools, particularly when they concern the use of computers, appears to be difficult to achieve in a consistent fashion across all of the experts involved in nuclear accident consultancy. When computer equipment is essential either for accessing information or for calculation work or for using the results, the setting up of a permanent team with good knowledge of the evaluation methods and the support tools can prove to be indispensable in order to assist the experts in their work and to ensure continuity of results from sampling work.
Of course these teams will be able to performs effective work only if staffing levels are assured right from the moment when the expert centre is set up.

6. Expert consultancy and information

Whether in the accident or post-accident phase, the objective-related time constraints and the necessity to employ special methods imply that the experts must acquire technical information in real time, that this can be interrogated locally (operator, state services), and that they are aware of the decisions made by the public authorities and the manner in which they have been implemented.

This also assumes that in each of the places from which information is obtained, the experts at national level will have designated contacts who are qualified, and who are freed from all operational tasks (decision or action), and that the managers of the expert groups should be in possession of suitable resources for transmission of their advice and recommendations to the decision-makers (telephone, facsimile, message service, computer cartography, etc).

During the accident phase, the information required by the experts will come from a limited number of sources (operator weather service, *préfecture*). Most of the technical information will be available on computer (the power station computer or the weather service computer) and can be transmitted by computer link. Other information will have to be sent by fax.

During the post-accident phase, however, since one of the main characteristics of information flow is the very large number of participants at national level, the communication resources must be able to cope with this. Use of computer communications and computer processing seems to be a good way to solve this problem.
The centralisation of measurements is imperative. Validation at national level is essential in order to be able to determine the feasibility of prohibitive counter-measures and on the necessity to introduce dispensations in the marketing area, on the need to get food supplies from abroad, etc.

7. Expert consultancy and communication

Adaptation of the communication network to suit the crisis organisation is vital of course.
In this organisation, one can identify:
- the Decision Centres,
- the Action Centres,
- the Expertise Centres.

If one assumes that each decision centre has action centres and expertise centres, then one can make rules for organising information transfer.
- The expertise centre should be able to assemble all information coming from its decision centre and the associated action centres.
- The expertise centres must be able to talk directly to each other, for the exchange of information and opinions.
- The expert centres will send out official advice only through the associated decision centre.

This information transfer scheme can only be effective if the equipment at each unit (expertise centres and decision centres) is as homogeneous, reliable and advanced as possible.
Only a network of special dedicated lines, dimensioned in order to ensure the correct exchange of information (speech, fax, etc), and reserved exclusiveli for use by the expertise organisation, will ensure an adequate degree of reliability.

8. Expert consultancy and training

It has already been emphasised that the experts must receive special training, to complete their area of knowledge, in order ensure that there is homogeneity in each of the expert domains concerned, and in order that a high level of expertise should exist in the management of serious accident situations and in appropriate methods of analysis.

It is obvious however that only periodic training will be possible in the places which will be uses in a time of crisis, with the tools made available, using exercises adding internal and external interfaces to allow them to prepare themselves in an effective manner.
Nevertheless, if one targets these exercises on certain parts of the system, this can result in effective testing and good awareness training - alarm testing, team-assembly testing, equipment testing, training in the use of equipment and methods, and so on.
National exercise, with the benefit of the experience gained from these tests, will just be more representative.
Experience shows that preparation of such exercises is very expensive in terms of people and time. The partial exercises are the more so for their necessary repetitiveness. Safety exercises involving operators and experts and simulating the accident phase take about two man-months of preparation and as much again for evaluation. National exercises which bring into play all of the state services, though without a great deal of deployment on the ground, require about one man-year of preparation, and the same for evaluation. Since the setting-up times for these are 2 months and 1 year respectively, the time required for analysis is much the same.
It can also be added that before calling in services such as installation safety, weather forecasting, situation evaluation, and so on, it will be necessary to have tested each of these areas by means of partial exercises, and that experts should have analysed all new interfaces in order to have learnt the appropriate lessons.

9. Conclusion

In order to deal with a crisis situation, it is not enough to install a suitable organisation. It is also necessary to have available groups of experts, professionals in the crisis situation, who are recognised as such by the operators and the state authorities and who are capable of supplying advice which is comprehensible to the public authorities.

Their expert advice will be dependent on specific evaluation methods which have been prepared and tested in cooperation with the operator and the relevant state services. In most cases, they will acquire the associated information in real time.

Expert consultancy cannot be provided effectively unless the necessary human and hardware resources at the different decision centres are selected, tested and installed beforehand.

Only exercises, with appropriate support scenarios to suit the set objectives, will enable the degree of preparedness for a crisis to be correctly assessed.

ORGANISATION IN CASE OF CRISIS IN A RESEARCH CENTER

MC TAMAS T PINEIRA

Centre d'Etudes Nucléaires de Fontenay-Aux-Roses
BP 6 - 92265 Fontenay-aux-Roses Cedex - France

SUMMARY

In case of crisis, the own means of establishment are put in activity according to Internal Energency Plan (in french PUI).

The present paper talks about the PUI of a Nuclear Center having only nuclear "laboratories".

It particulary develops :

- complementarity with exteriory means actions (such us fireman...)
- environment first controls : organization, means, etc... being incumbent on establishment.

INTRODUCTION

France has defined, for her nuclear industry, a very strict preventive organisation ;so everybody, operators and safety authorities, has to take care and keep out of any type accident.

Nevertheless, each poblematic case is systematiccaly analyzed and French Nuclear Research Center get arrangments wich are collected in an internal emergency program (French denomination for this is being "PUI" wich means "Plan d'Urgence Interne").

"PUI" term is common in every nuclear installations, whatever their importance and activities may be. Eventhough all these electro nucleary power stations look very much alike, the Research Centers may defer from one to another. Any "PUI" exposes what to do in case of crisis but the main poins defining the way to applicated the plan ought to differ.

PRESENTATION OF "CENFAR"

Situation

The Nuclear Research Center in Fontenay-aux-Roses, "CENFAR" is built in two parts ("Fort" and "Annexe") separated by a public way and stood on a 13,8 ha surface, in Fontenay-aux-Roses urban area. It is some one hundred meters away for 3 other towns : Clamart, Chatillon-sous-Bagneux and Le Plessis-Robinson, and also about 1 km away from Bagneux, Chatenay-Malabry and Sceaux.(s.1)

Also really lose to Paris Notre-Dame (9Km), the center is 3,4 Km distant from Paris ring-road.

This reduced area gathers 70 buildings of different importance, 4 of them are Basic Nuclear Installations "INB". Because ot the reduced area, some of buildings, even the "INB" ones, are quite near of "CENFAR"'s limit and of the suburd habitations.

Activities

The most important INB regroups mainly chemistery laboratories, which concentrate their work on research, including work on reprocessing -only for "cooled" combustibles-, analyses and manufacture of transuranium sources.

The 2 other INB collect, condition, store and process radioactive wastes.

The fourth and last one is being cleansing before dismantling.

NB : there has been no more nuclear reactor in the "CENFAR" for years

CRISIS SITUATION

Hypothetic accidents

Studies of accidents, eventhough they are hypothetic and very unlikely, have shown that the less negligible probability would be gaz or aerosol dispertion outside of the buildings, due to a fire on a large part of the "INB". The eventual effects would not spread over more than 1 km around the area, whatever the importance of the accident, but the mediatric repercussions would be immediate.

To note than "criticity", even though it has been considered, would carried a minor emission in the environnement, because of the small quantity of product concerned.

Particularities and important points

Analyzing an hypothetical crisis situation shows some contradictory points at first sight, but needing to be considered.

They are :

. the materiel consequences (people or goods) of an accident would be very small.

. a dispersion out of a building can extend over the environnement very quickly given the size of the CENFAR, but the quantity would be also very small.

. the neighbourought would immediatly be warned, even there is only appearence of abnormal situation.

. therefore media would be also warned.

. mediatic consequences could be very important.

INTERNAL EMERGENCY PROGRAMME -PUI

Conception and aplication of the "PUI"

Refering to the rules wich shall be applicated, "PUI" is :

. a descriptive document (organisation in a normal situation, in case of crisis means...),

. a practical document for people or units thats need to interfere. This part of the document is made of "reflex slips".

Refering to the "CENFAR" and given what was explained before (urban situation and size of the center), an abnormal situation would impose very fast activation of "PUI", and would just as fast involve the environment, which even would imply post accidental actions.

In any this case, preventing these series is not mainly the result of maintaining the installation in "safe leading", even then this situation is starting, but depends overall on the efficiency and rapidity of specialized units. In fact, in case of fire the "safe situation" is the folling back, of course avoiding secondary accidents, wich is under the responsability of the installation's chief, but, above all, the firemen have to limit its spreading and subdue it.

Besides, even if the fire is subdued very quickly, the interference itself causes damage to the building's "barriers", just only due to comings and goings of the security groups (firemen, healthphysicmen...)

It is, then, necessary to control the environment without any delay.

That's why :

. first, "PUI" set itself going progressevely (while than word "declenchement" seams to be a sudden modification in men's actions).

. then, the role of the Local Security Group (FLS) -firemen-, of the Healthpysic Unit, particularly of the Environnement and area Radioprotection Section (SRSE) is to be underlined.

Consequently, not only the reflex slips are divided in units and types of interventions, situations, contrôls... but they also applie to any abnormal situation. If anything (telltal sign, person) witnesses the presomption of an accident, "FLS" would immediatly warns operating units, which would follow their own rules, exposed on the reflex-slips.

Three different persons can decide whether to establish a Local Command Post "PCL" depending on the importance, the nature and the evolution of the situation : installation's chief, FLS's chief and healthphysic unit's chief. In the same way, the Center's Director decides whether to create the Local Direction Command Post "PCDL" which means activation of the "PUI".

Specialized units are acting by themselves while the "PUI" is not activated but, also, they are the knot of different staffs: Control Staff "EC", Crisis Technical Staff "ETC", Movement Staff "EM" who becomes official at the same time than the "PUI"'. However the task of the personnels do not change.

"ECT"'s role is to help the installation's leader on technical points (ventilation, distribution network..)

"EM" takes care of logistics.

"EC", which is then the new name of "SRSE", has a prime role : it must collect and interpret all the radiological and meteorological measures.

This organisation could be symbolized with a triangle; each top would be "EC, PCL and PCDL" (s 2).

The datas collected are transmitted from "PCL" to "EC" and back, and then, the two to "PCDL" the redondance is another security for transmission.

Activation of outside means in case of fire

In case of fire, the first unit concerned is the specialized unit, which may also need outside assistance. This decision has to be taker by FLS's chief, following the "PISE" (outside assistance intervention programe). This assistance is given by "Paris Firemen Brigade - BSPP" located in Plessis-Clamart.

It is to be noticed that "BSPP" intervenes inside the installations where the Center's Director assumes the responsability of actions, and of their consequences, but else, the mobile group for radiological intervention "CMIR" is constitued by a part of the BSPP'members and it can be activated in environment, on the Prefect's decision, in case of Intervention Particular Program "PPI". Moreover, in this case, the Prefect may decide to put the General (BSPP'chief) or his representative at the head of the Advanced Command Post "PCA".

Ways to act on the CEN-FAR'area and for the environment

Six control stations transmit their datas to the environment control board "TCE" (s 1) in real time, under the control of the "SRSE", as well as 4 light vehicles, equiped, or canning be equiped, which communication, measurement and sampling tools. One of the cars can also analyzes the samples.

Light vehicules were prefered to heavier ones, with elaborated means, because, given the urban trafic in this area and the limited concerned surface, they can carry easily the datas back to be analyzed in labs, either to the center, or to Saclay center if labs'"CENFAR" are inaccessible (Saclay is only 15 Km distant from FAR).

"SRSE"'s leader decides whether to use those vehicules as soon as there is a doubt about a rejection or a dispersion on the site, therefor in the environment, following pre-established advance, given the meteorological conditions - mainly the direction of the wind. There are 18 different routes around the center (according to sectors distant of 20° each) on which are located particular places. To each place corresponds a serie of measures and samples to do, and a code for transmission to the lab, to prevent confusion while collecting and analyzing the datas. As soon as the "PUI" is activated, the tasks of each group cannot change; they can only be ordered to come back in case of false alarme. However, they may be modified in case of PPI; Prefect, having the responsability of actions on environment, may, then, give other directions

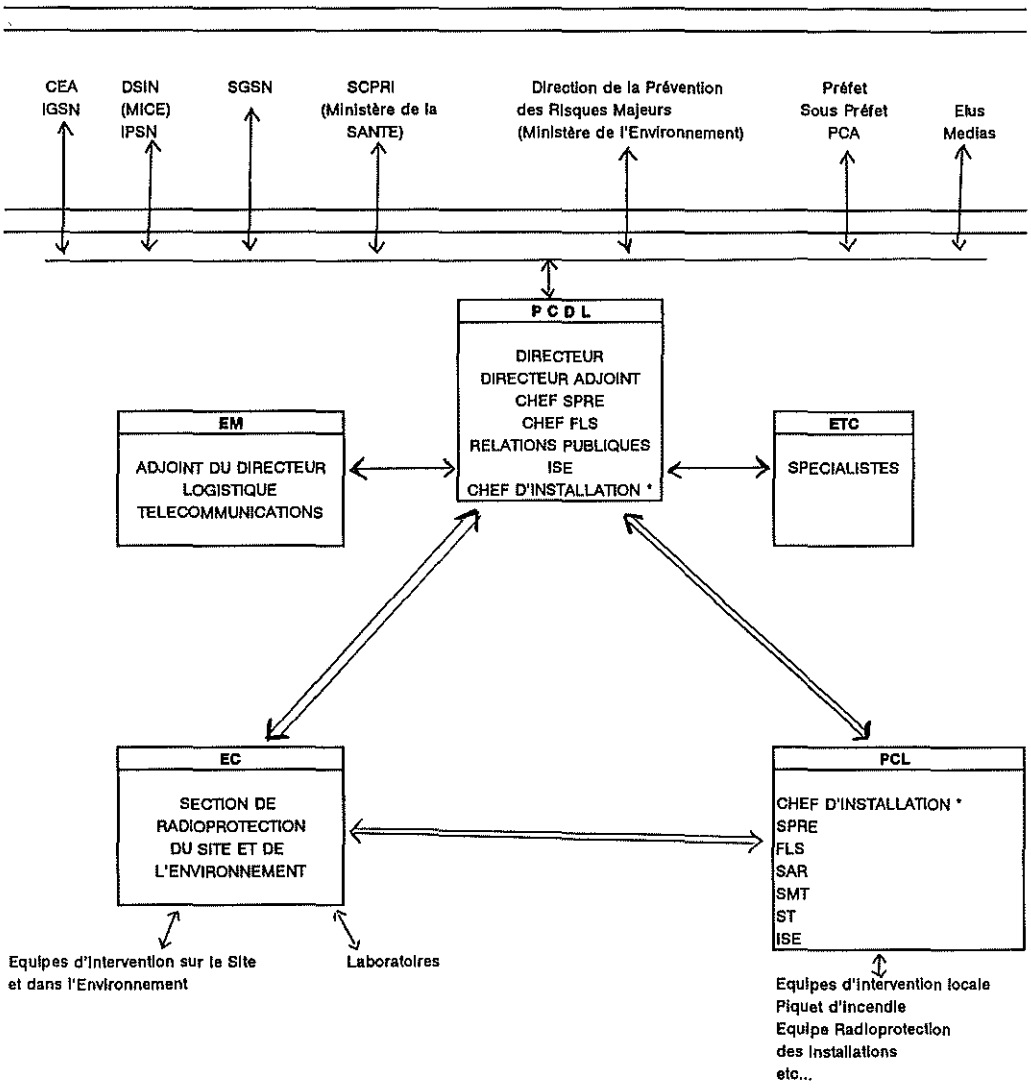
This activation, which may seen early is due to the location of the "CENFAR". It is obviously necessary to give fast informations, to evaluate the consequences as close as possible of the reality. It has been shown, after having studied all the possible cases of accident, that it would never be necessary to evacuate even part of the population. But it is to be prouved such as accident will to stay in the forcasted limits, ans then to convince responsible authorities for the final decision.

CONCLUSION

"CENFAR's PUI" is made to answer to the specific needs ot the center. Consequently, it is and it has to be, different from the electronuclear power stations of EDF or from some other research centers of CEA, or COGEMA's establishments.

It is then important that the rules to their elaboration can be adapted to each case, and not a real transcription.

ORGANISATION EN CAS DE CRISE



SPRE : Service de Protection contre les Rayonnements et de L'environnement
FLS : Formation Locale de Sécurité
SAR : Section d'Assainissement Radioactif
SMT : Service Medical du Travail
ST : Service Technique
ISE : Ingénieur de Sécurité d'Etablissement

DSIN : Direction de la Sûreté des Installations Nucléaires
IPSN : Institut de Protection et de Sûreté Nucléaire
SGSN : Secrétariat Général pour la Sûreté Nucléaire
SCPRI : Service Central de Protection contre les Rayonnements Ionisants

PCDL : Poste de Commandement de Direction Local
PCL : Poste de Commandement Local
EC : Equipe Contrôle
EM : Equipe Mouvement
ETC : Equipe Technique de Crise
PCA : Poste de Commandement Avance

MONITORING RADIONUCLIDE EMISSIONS DURING CONTAINMENT VENTING

R. Hock, G. Röbig
Power Generation Group (KWU) of Siemens AG, Depts. R26 and R262,
Berliner Strasse 295-303, W-6050 Offenbach am Main,
Federal Republic of Germany

Summary

At the request of the Technische Vereinigung der Großkraftwerksbetreiber (VGB), an association of large power plant operators, Siemens has developed a system for monitoring and documenting the emission of radionuclides in the event of a beyond-design accident in which pressure buildup in the containment necessitates containment venting. The monitoring system complies with the recommendation made on November 4, 1987, by the German Commission for Radiological Protection (SSK) and has since received positive evaluation. The paper describes the requirements to be met by the monitoring system, its design and operation, as well as the current stage in system planning and installation at German nuclear power plants.

1. Introduction

In response to recommendations made by the German Reactor Safety Commission (RSK), all light water reactor (LWR) nuclear power plants presently in operation in Germany either have been or are presently in the process of being backfitted with filtered containment venting systems. The purpose of these systems is to limit pressure buildup in the containment in the event of a beyond-design accident by venting air and steam from the containment to the plant environs through a filter system.

The new emissions monitoring system provides nuclide-specific monitoring of radioactive materials discharged during filtered containment venting, broken down into the following nuclide groups:

- Aerosol-bound radionuclides
- Non-aerosol-bound (gaseous) iodine isotopes, and
- Radioactive noble gases.

The data on emission levels determined in the event of containment venting are mainly intended to serve as a basis for implementing emergency response actions and as accurate accident documentation.

2. Monitoring System Requirements

The gas flow to be monitored is a mixture of air, water vapor and other gases, the composition of which may or may not change considerably in the course of venting. Pressure, temperature and relative humidity of the monitored gas flow depend on

whether samples are taken from the venting system discharge line or downstream of the venting system inlet into the stack after the venting system exhaust air has mixed with the plant exhaust air flow (Table 1). A main factor governing the design of the monitoring system is the maximum activity concentration occurring in the vent gas flow in the initial phase subsequent to initiation of containment venting during a core melt accident. Table 2 shows the design requirements in this regard determined on the basis of the RSK recommendations of May 30, 1990 (Appendix 1 of Report of the 253rd Meeting of the RSK) for two types of plants, one equipped with a pressurized water reactor (PWR) and the other with a boiling water reactor (BWR).

With respect to laboratory evaluation, activity loading of aerosol and iodine filters should be limited to 1E9 Bq. This level allows measurements to be performed in the nuclear plant's on-site laboratory using standard techniques.

The monitoring system should be controlled from the main plant control room, and should be designed for sufficient measuring accuracy: deviation of measured values from actual values should not exceed a factor of 3.

Table 1: Conventional Design Requirements

Application	pressure in bar	temperature in °C	relative humidity in %
BWR containment venting	1 - 2	approx. 130	< 100
PWR containment venting	1 - 2	approx. 130	< 100
PWR plant exhaust air	1	50 - 80	100

Table 2: Design Requirements with Respect to Concentration of Radioactivity (Bq/cbm)

Application	aerosol-bound radionuclides	non-aerosol-bound iodine isotopes	radioactive noble gases
BWR containment venting	1E11	3E11	8E14
PWR containment venting	4E7	7E9	1E13
PWR plant exhaust air	6E6	1E9	2E12

3. Design and Operation of Monitoring System

The discharge of radioactive materials during containment venting is monitored using a gamma dose rate monitor, an aerosol monitor and an iodine monitor, as well as an aerosol and iodine sampler (see Fig. 1).

In BWR plants, the gamma dose rate monitor is installed on the venting system discharge line; this monitor measures the total activity release rate, which is mainly determined by the noble gas release rate. Since it is assumed that containment venting at a PWR plant would not be initiated until several days after reactor trip (in contrast to BWRs, where venting could begin after four hours), the gamma dose rate monitor at a PWR plant is located downstream of the aerosol and iodine sampler owing to the low-energy gamma radiation of the radioactive noble gases; it therefore serves to monitor noble gases only.

The release rate of aerosol-bound radionuclides is monitored using an aerosol monitor, and that of non-aerosol-bound iodine compounds by means of an iodine monitor.

For control and backup purposes for these monitors, an additional bypass is also routed via aerosol- and iodine filters (sampler). The radioactive substances accumulated over the entire venting period can be measured after the event. Due to the fact that flow in this train is continuous and at a higher rate the accumulated acitvity can be such, that it may be required to either allow some decay time, or to evaluate the sample in a specialized laboratory.

The monitors and the sampler, located in an appropriate compartment, are supplied with their gas flows via a bypass system. The compartment is selected to achieve the shortest possible sampling line length and to minimize the effects of background radiation on the monitors. Also taken into consideration in the selection of this location is the radiation exposure of operating personnel when a loaded filter has to be transferred to the plant laboratory for nuclide composition analysis.

4. Important Features of Monitoring Concept

a. The sampling line system and the connected monitors and sampler are thermally traced to prevent condensation and thus reduce deposits. The sampling head is also heated in venting systems in which the gas flow is supersaturated with water vapor in order to ensure rapid droplet evaporation.

b. The sampling line system design has been optimized to reduce deposition of aerosol particles and elemental iodine (e.g. in the case of elemental iodine through the provision of Teflon surfaces).

c. Aerosol and iodine monitor throughput is low and residence time is selectable, thus allowing samples to be obtained which can be safely handled and evaluated in the laboratory.

d. Noble gases are purged prior to sample transport and analysis, thereby facilitating handling and significantly increasing measuring accuracy.

e. Samples (i.e. the filter material from the aerosol and iodine monitors) are transferred manually to the radiochemical laboratory for precise determination of nuclide-specific releases according to the adsorption capabilities of the individual chemical compounds discharged (aerosol-bound radionuclides, elemental iodine and organic iodine); the analytical techniques to be employed under accident conditions are practiced during normal plant operation in order to reduce the stress factor for personnel.

f. Monitoring system operation is controlled from the control room via a programmable control system housed in a control equipment cabinet located at the sampler and monitors.

5. Supplementary Information on Aerosol and Iodine Monitors

In the aerosol monitor, the aerosol-bound radionuclides are retained on a fiberglass HEPA filter and monitored by an NaI(Tl) detector, which provides for integral measurement of the gamma radiation emitted from the filter. The detector signals are processed by the associated transducer to obtain analog output signals for "aerosol filter loading" and "aerosol concentration" which are displayed and recorded in the main control room.

The iodine monitor is similar in design to the aerosol monitor. A special combined filter unit (dual-bed filter) is employed for iodine removal. In the first filter bed, elemental iodine is retained by an adsorbent impregnated with potassium iodide, while the second filter bed removes organic iodine by means of an adsorbent impregnated with silver nitrate, thus allowing elemental and organic iodine to be separately measured during laboratory evaluation.

6. Planning and Backfitting of Emissions Monitoring Systems

To date, this emissions monitoring system has been installed and is operational at one twin-unit LWR nuclear power plant in Germany.

Emissions monitoring systems have been ordered for three other LWR plants and Siemens is in the process of planning emissions monitoring systems for another ten LWR plants.

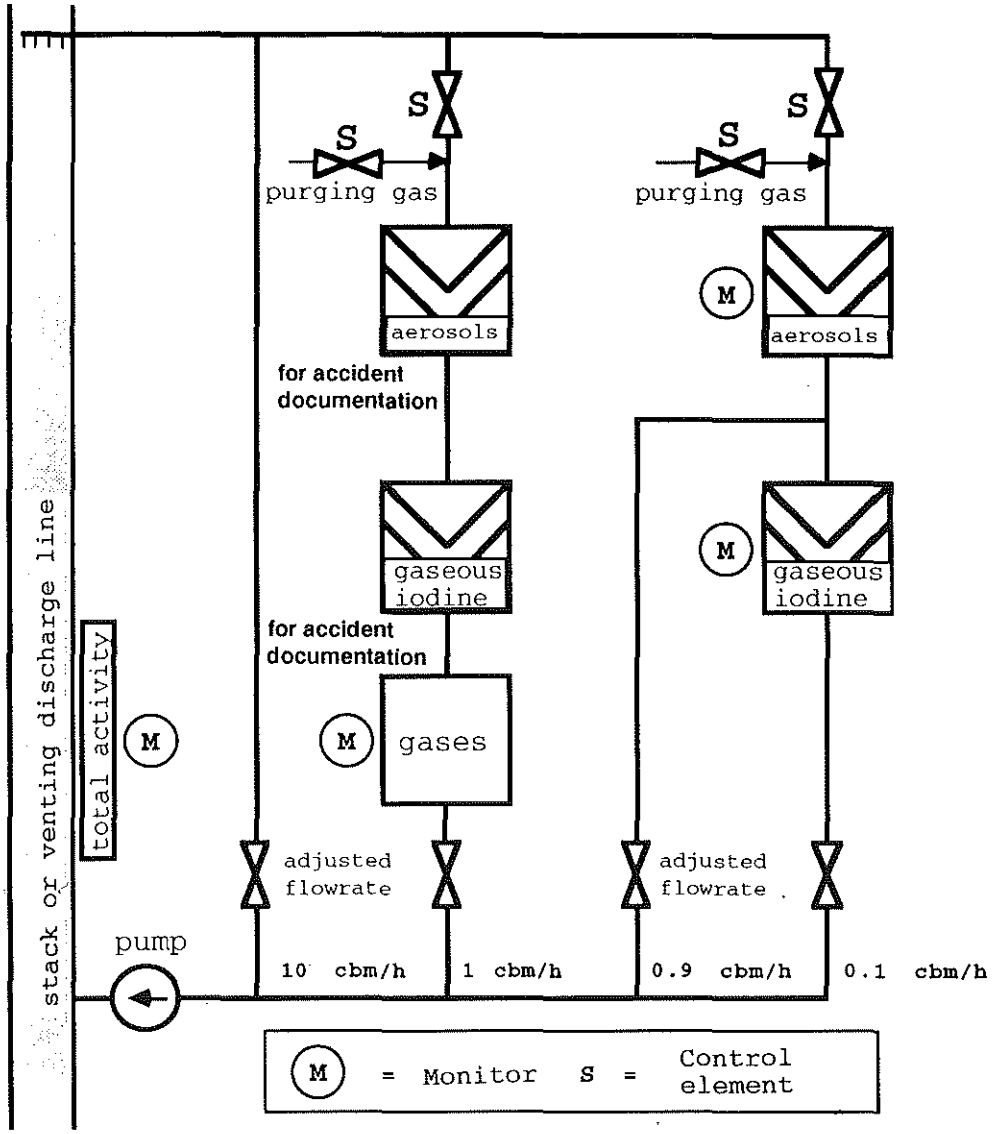


Figure 1: System for Monitoring Releases of Radioactive Substances to the Environment During Accidents

RADIATION MEASUREMENT VEHICLES AND RADIOTRANSMISSION OF DATA OF THE EMERGENCY SERVICE GROUP KHG

H.Hauske, F.Stettner, Kerntechnische Hilfsdienst Gesellschaft mbH,
D-7514 Eggenstein-Leopoldshafen 2

Summary

KHG maintains staff and equipment in order to assist the operators of nuclear facilities in case of an incident or an accident. In the frame of this tasks KHG runs three radiation measurement vehicles which are equipped with all necessary devices for sampling, measurements and communication to carry out the environmental emergency measurements near a nuclear facility.

The measured radiological data are transmitted by radio in form of a protocol to a radio lead vehicle. The data are stored in a computer. A hard copy of these data is sent as telecopy to the respective plant operator. This procedure avoids transmission failures and communication mistakes and decreases voice communication.

To improve the data transfer the reception antenna will be increased and an additional data transfer system using mobile telephones will be established in the next future.

1. Introduction

On the base of Section 38 of the German Radiological Protection Ordinance KHG maintains staff and equipment in order to assist the operators of nuclear facilities in case of an incident or an accident. The main objects of KHG are radiation protection, decontamination and teleoperation. An important task in the frame of these objects is to carry out the environmental emergency measurements near the plant which are fixed in German regulations (Fig.1).

FIG. 1:
Accident training measurements and telecopy form

(Nuclear power plants)

Gamma dose
Gamma dose rate
Aerosol activity (total beta)
I-131 Activity in air
Soil surface activity (total beta)

Telecopy from KHG

To .....

Time : .....

Place : .....
Date : .....
Time : .....

Wheather : .....
Crew : .....

Dose rate (µSv/h) : .....
Soil contam. (Bq/cm²) : .....
Plant contam. (Bq/cm²) : .....
Aerosol activity (Bq/m³) : .....
I-131 Activity (Bq/m³) : .....

For this job KHG runs three radiation measurement vehicles which are equipped with all necessary devices for sampling, measurements and communication including radiotransmission of data.

For communication with KHG crews and the plant operator KHG holds a special equipped radio lead vehicle.

2. Radiation Measurement Vehicle

The newest of the three radiation measurement vehicles is a Mercedes Benz DB 710 D with a Diesel engine of 70 kW (100 PS) and an optional four wheel drive. It is equipped with four seats and its total weight is 3.7 tons. The ceiling's height inside is about 2 m in order to enable the crew to stand inside the car during work.

The sampling equipment contains

- 2 aerosol samplers (flow rate 55 m^3/h),
- 2 Iodine samplers (flow rate 6 m^3/h),
- sampling devices for soil, plants and water.

For radiation measurements the vehicle is equipped with

- a low dose detector FHZ 600 A
  (measuring range 0.01μSv/h - 1 mSv/h),

- various dose rate measuring devices,
  * 6150 AD2 (0.5 μSv/h - 10 mSv/h),
  * 6159 ADT (5 μSv/h - 10 mSv/h),
  * Babyline 31 (0.2 μSv/h - 1 Sv/h),
  * Szintomat (0.1 μSv/h - 0.1 Sv/h),

- a shielded aerosol filter detector FH 650 E (15 mm Pb-shielding)
  with evaluation device FHT 7000,

- 2 contamination monitors FHT 111F,

- a portable multichannel analyzer with shielded high purity
  Ge-detector (40 mm Pb-shielding, efficiency 25 %),
  connected with laptop and printer for data evaluation.

All measuring devices are battery powered. The samplers are operated by a power generator (1.9 kVA, 220 V AC) because of the high powerconsumption of these devices.

3. Communication and Radiotransmission of Data

For communication with the radio lead vehicle and others the radiation measurement vehicles are equipped with radio and portable telephone. For transmission of data the radio devices are connected with a laptop including a special radio modem.

The radiological data collected in the environment by the measuring crew are input in a computer form and checked on plausibility by the input program. The data are sent to the

radio lead vehicle by a special transmission program (Fig.2). This program checks the transmission for failures and repeats automatically the transmission as required.

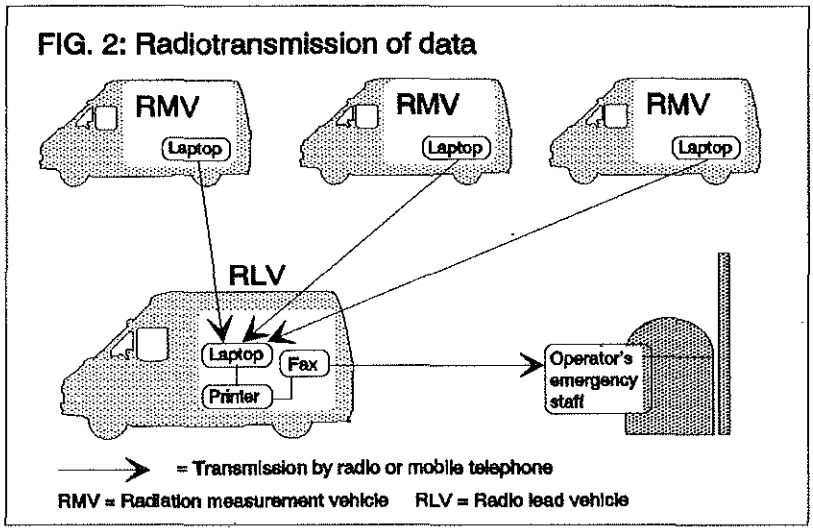


FIG. 2: Radiotransmission of data

The sent data are collected in the laptop of the radio lead vehicle. A telecopy form equal to the input form (Fig. 1) is printed of each radio protocol. This hard copy is used for documentation and is telecopied to the operator's emergency staff by a telecopier connected with the mobile telephone.

The advantages of radiotransmission of data are that transmission failures cannot occur, communication mistakes are avoided and voice communication is decreased.

The main problems of the existing radio system are the transmission range (only about 10 km in flat land) and the occurance of radio shadows by e.g. hills and buildings. To improve the data transfer the reception antenna will be increased and an additional data transfer system using mobile telephones will be established.

WIDE CONTAMINATED AREA CHARACTERISATION

André BERTEL, Commissariat à l'Energie Atomique, Direction des Applications Militaires, Centre d'Etude de Ripault, BP 16, F-37260 MONTS.

The characterization of a widely contaminated zone, leads to two questions :

- which organization to choose ?
- which implementation means can be used ?

We are going to briefly cover these two points, limiting them to a local level.

Which organization to choose ?

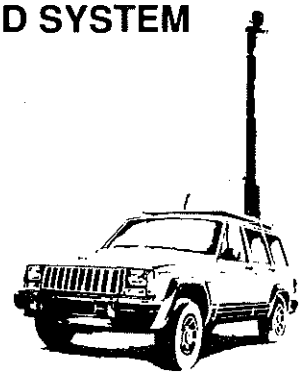
This revolves around three entities :

- the CEA advisor to the head of country ,
- the advisory group who receives all experts concerned in the surrounding agricultural produce, prepares the replies to questions asked by the civilian authority and directs the work on the field.
- the CEA advanced command post who ensures the characterization of the area according to the needs of the advisory group (see figure "advanced command post").

Which means can be used ?

- the first information comes from the fall out prevision centre who then directs the searches.
- the exploration of the real contaminated zone is then the first concern of the civilian authorities. So, the use of the GAMMA detection by air allows for the authorities to inform rapidly the extent and activity of the zone and this is carried out by an helicopter which covers 1000 acres by flight.
- the detection on the ground as well as the sampling at sensitive points are ensured by the mobile teams in vehicles equiped with detection means. The samples will be sent to diverses national laboratories.
- finally, the non-contamination control of zones is carried out by vehicles such as the VISA, with atmospheric control.

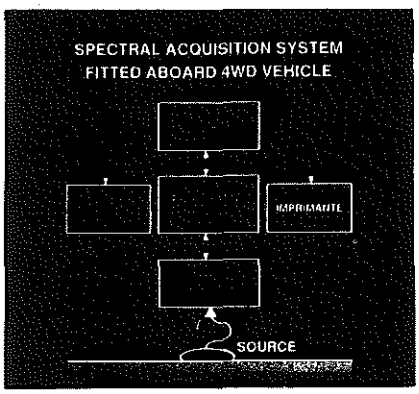
GAMMA-RAY SPECTROMETRY FIELD SYSTEM



GAMMA-RAY SPECTROMETRY FIELD SYSTEM carried by 4WD vehicle for routine or accident checks
ides accurate identification of natural and artificial radionuclides and swift measurement of the levels of
activity over an area of several tens to several hundreds of square metres. It is composed of a liquid
gen cooled Germanium (HP) detector and a multichannel analyser. Detector height from the ground can
djusted by a telescopic mast which folds down onto the vehicle roof for transport.
indispensible back up to AIRBORNE GAMMA CARTOGRAPHY.

FUNCTIONS

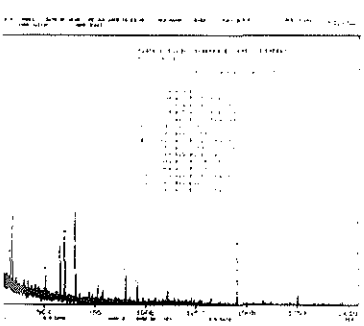
- Accurately detects and identifies radioelements (g spectrometry)
- Detects and quantifies surface activity (expressed in Bq/m²)

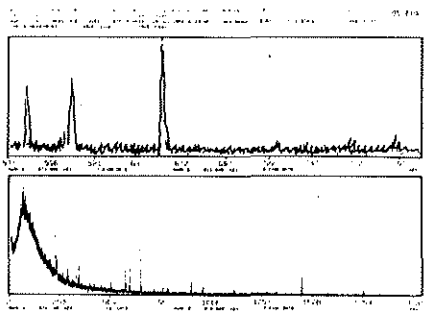


PERFORMANCES

- Multi-channel analyser : 1024 to 8192 channels
- Detector battery life : 100 hours
- Height of detector from ground : 2.50 m - 8.00 m
- Analysis results : 1 hour
- Levels of sensitivity :(T=1800 s H=2,50m à 3,50m)

E ≃ 100 KeV: 0,2 KBq. m⁻²
E ≃ 500 KeV: 0,1 KBq. m⁻²
E ≃1000 KeV ≤ 0,1 KBq. m⁻²





SIMULATION OF A NUCLEAR ACCIDENT, GATHERING AND TREATMENT OF ENVIRONMENTAL DATA DURING THE *CADARACHE DRILL* OF OCTOBER 1991

G. BOUSQUET-CLAYEUX*, S. DESCOURS**, Ph. KISSEL*, A. L'HOMME*
D. MANESSE*, J. MATUTANO***, Ph. PICAT*, D. ROBEAU*,
M. WILD****

* Institut de Protection et de Sûrete Nucléaire, Département de Protection des Installations et de l'Environnement - Fontenay-aux-Roses (France)
** Centre d'Etudes de Grenoble / Direction - Grenoble (France)
*** Direction Centrale de la Sécurité - CEA - Paris
**** Direction Régionale de l'Agriculture et de la Fôret - Marseille (France)

SUMMARY

In October 1991, the CEA Group organized a drill to check its capacity to characterize large contaminated zones. Before this drill, samples have been marked with radionuclides, during the drill large means of measurement and of sampling have been deployed in the country and many analysis laboratories were on duty. According to a plan done from the forecast fall-out, actual samples have been sent, with a checking form giving the simulated contamination to the various laboratories. All the results have been transmitted on the SYTAR network and were available in all the crisis centres. SYTAR is a remote permanent network linking the radiation protection services of the CEA installations.

1 - INTRODUCTION

The french national emergency plan, applicable in case of a nuclear accident, provides the intervention of the human and technical means of the CEA and COGEMA working with the local authorities, the operator and the Ministry of the Interior, the Ministry of Health and other ministries (Agriculture, Economy, Defense, etc...).
In October 1991 a nuclear safety exercise concerning the post accident phase was organized by the IPSN (Institute of Nuclear Protection and Safety) with the aim of demonstrating in actual size the means and methods that might be used for :

- delimitating the extended contaminated areas and characterizing them from the point of view of radiological risk,
- organizing the collection of samples (water, vegetable and animals products) in the field,
- controlling entries and exits of personnel and vehicles into and out of the contaminated area,
- monitoring, transferring, and analyzing the samples taken,
- interpreting the results and presenting them in synthetic form to the authorities,
- advising the authorities and participating in decision making.

To carry out all these duties, the CEA group works with administration and with territorial government means of intervention. This exercise took place during four days on the Cadarache site ; it involved about 150 persons. In addition to observers from the central administrations involved, who followed this operation (first of this type on the national level), experts were charged with observing the operations and supplying reports on specific themes.
A video team and professional journalists under contract produced a journal in real time and a film tracing the salient facts of the exercise.

2 - PREPARATION FOR THE EXERCISE

Preparation for the exercise was entrusted to the interveners themselves and to a coordination and scenario writer team.
A detailed scenario was given to the actors before the operation to make it possible to begin the exercise ; it included an accident scenario and a population scenario.

Accident scenario :
This part introduced succinctly the installation, the conditions of the accident, an estimate of the weather conditions, and simulated releases (on the order of 1.10^{-4} of the inventory of a 900 MWe PWR core for the most volatile products : I, Cs, Te). East winds blowing caused contamination towards the lower valley of the Durance.

Population scenario :
Countermeasures alleged to have been put in place involved :
- evacuation of the nearest populations with lodging in cities located out of the passage of the plume,
- sheltering in houses up to a distance of about 10 km,
- prohibition of traffic on some roads, highways and railroads,
- suspension of local markets,
- closing of schools,
- recommendations not to eat fresh local products (green vegetables, fruits, milk, etc...).

A video tape was shown at the beginning of the exercise to remind all the actors of the elements of the scenario at the same time.
Environmental contamination was simulated in two ways for the exercise :
- by putting into circulation 130 samples marked with I131 and Cs137 beforehand,
- by the use of the GEREM software; this software that can be used on portable microcomputers utilizes SIROCCO code results (IPSN code to evaluate accidental fall-out) and translates them into terms of contamination of air, soils and food.

To prepare the work of the CEA experts and the governmental departments during the exercise, significant documentation was collected on the state of agriculture, livestock, and the water in the region (general agricultural inventory, cultural calendars, marketing flow, water networks, etc...). Finally, the agricultural producers and stock breeders were informed by the local administration that intervention teams would come to ask them for samples of their products. In addition, the Cadarache Center management published a press release

announcing the exercise, and the persons who were going in the field could distribute an information flyer to facilitate contact with the population.

3 - REALIZATION OF THE EXERCISE

Chronology of the exercise (October, 1991)

* *Sunday 13* (outside the exercise, not played by actors)
- accident at a fictitious installation,
- launching of the Internal Emergency Plan and of the Particular Intervention Plan

* *Monday 14*
- implementation of the plan of action,
- movement of the mobile teams towards the Cadarache Centre,
- arming the command posts (CP),
- welcoming the teams, presentation (with video film) of the exercise and of the role of each team,
- estimate of the consequences by the IPSN Emergency Technical Center (ETC),
- preparation of the teams' duties.

* *Tuesday 15 and Wednesday 16*
- reworked estimate of the consequences by the ETC, taking into account new information,
- field manoeuvres of mobile teams and those transported by helicopter ; taking into account the imaginary level of contamination, the area explored covered about 150 km^2,
- transfers of samples to the laboratories,
- interpretation of the first results.

* *Thursday 17*
- follow-up of the laboratory work,
- completion and conclusions of work in the CP,
- departure of field teams to their centres,
- disarming the CP,
- general debriefing.

The progress of the exercise required the participation of numerous actors both on the local level and on the central level (Paris). These actors were grouped into different operational stations.

3.1. - Préfet command station

The command station of a Préfecture was simulated at the house of the Cadarache hosts. The Sous-Préfet of Aix, assisted by representatives of the departmental or regional offices of civil protection, agriculture, and social matters of Bouches-du-Rhône and Vaucluse, played the role of Préfet, local authority, head of the post-accident plan. Supported by an advisory unit that grouped specialists (in radiation protection, transfers into the environment, cartography, and agriculture and food problems) a CEA advisor to the Préfet was in charge of relations between the Préfet and the operational teams. As such, on one hand he had to present a synthesis of the results obtained in the field regularly to the Préfet and, on the other, to direct operations through the head of the advanced CP and inform him of the Préfet's requests.

3.2. - CEA advanced command station

The head of this station, located near the contaminated area was responsible for :

- organizing the missions of the teams in the field,
- controlling the entries and exits of teams into and out of the contaminated area (radiation protection of the interveners, decontamination of persons and vehicles),
- posting and routing the samples taken to the analytical laboratories,
- simulating the results of contamination for unmarked samples,
- transmitting the simulated results of direct measurements in the environment (dose rates, gamma spectrum) to the advisory unit to the Préfet and to the CEA Center (CCC) and Emergency Technical Center (ETC) of the IPSN.

To carry out these duties the head of the advanced command station had the following means at his disposal:

- five light intervention teams, equipped with vehicles, detection material, and UHF transmission devices,
- two heavier intervention teams, equipped with an atmospheric control vehicle (DASA and VISA 90),
- a light team, the so-called Mobile Radiological Intervention Cell of the Vaucluse Civil Security Administration,
- a team transported by helicopter and a ground team from the economic interest group "Automated Accident Intervention" (INTRA) which unites EDF, COGEMA, and the CEA,
- a vehicle called GEMINI, conceived and made by the Central Protection Department against Ionizing Radiation to measure the internal contamination of individuals,
- a team from the Radiation Protection Assistance Office of the IPSN in charge of controlling the samples and transmitting them to the analysis laboratories,
- a scenario team responsible for associating a simulated contamination value with each measurement in the environment and each sample taken,
- a team responsible for entering the results of measurements in the environment on the SYTAR information processor.

3.3. - Analytical laboratories of the CEA and COGEMA centers

Five CEA and COGEMA laboratories and one laboratory Marseilles Veterinary Services laboratory received and analyzed the samples (about 20 per laboratory) marked with I131 and Cs137 beforehand.

They also received samples taken in the environment accompanied by an envelope indicating the simulated levels of activity. After the samples of the first type were counted, the personnel had to transmit the calculation results and the simulated contamination represented in the envelopes through the SYTAR information processor to the ETC and the CEA advisory unit.

3.4 - Emergency Technical Centre of IPSN

Taking into account the information available to it (source term, weather), on the first day of the exercise the ETC had to draw up estimated maps of the environmental contaminations to direct field operations.

Secondly, as soon as information arrived on the SYTAR processor or by fax, it had to draw up a realistic map of the contaminated areas and construct a data bank concerning the levels of activity in the main products of the food chain. This work was done in conjunction with the advisory unit, which on

its part organized an assessment of the contamination in order to inform the Préfet and suggest countermeasures.

4 - CONCLUSION

The conclusions are drawn from the basic notes made by the interveners and the reporters.
Operational plan of action :
The entire plan of action turned out satisfactorily ; however, it should be noted that the case treated was simplified in relation to a real case, which would involve numerous administrations, socio-economic agencies, the medias, the population, and the political class.
The roles of advisor to the Préfet, specialists (for the advisory unit and the ETC), and Head of the advanced CP are very important and require training and the involvement of a sufficient number of persons capable of filling these functions.
Means used in the field :
For external interventions during the exercise, the difficulties encountered showed that :

- measurement materials should be robust, have direct reading, and cover a large measurement range,
- vehicles should be all-terrain, equipped with an external device that makes possible continuous measurement ; they should be sealed from contamination or adjusted to facilitate decontamination; on the other hand, it is necessary to provide high-performance radio connections and a radiotelephone.

A convention for taking samples must be established, and the material must be standardized. The importance of the country laboratory, which makes it possible to obtain the first results (gamma spectrum to characterize the main radionuclides) should be noted. This laboratory should make it possible to sample material from the field teams and unblock the laboratories by conducting a first sorting of the samples.
Laboratories and data circulation :
The analytical laboratories of the radiation protection services of the CEA group centres were well equipped to answer the demand ; the results of the activities measured agreed well with the values marked beforehand.
It is suitable to note the significance of:

- standardization in packing, which should be adapted to the measurement stations,
- the labelling and presence of a well informed follow-up file (exact type of sample, coordinates, and time of this sample).

CONNECTION BETWEEN THE IMMERSION-DOSE RATE IN THE ENVIRONMENT AND THE MEASUREMENTS OF THE DOSE RATE IN THE CONTAINMENT AND THE STACK AFTER AN ACCIDENT

Franz Cartier, Andreas Leupin
Swiss Federal Nuclear Safety Inspectorate
CH-5232 Villigen-HSK, Switzerland

Summary

In the environment of a NPP, the dose-rate D_H resulting from immersion in the cloud (mainly by noble gases) is the dominant factor in the early phase of an accident with radioactive release. A simple and fast estimation of the gamma-dose rate D_H may be obtained from the respective dose-rates D_S and D_C measured by the wide-range-ionisation chambers in the stack (S) and the containment (C) using the formulas

$D_H = \chi \cdot V_S \cdot GF_{S,E} \cdot D_S$ and $D_H = \chi \cdot V_C \cdot (p/p_C) \cdot GF_{C,E} \cdot D_C$. {eq. 1, 2}

In these formulas χ corresponds to the dispersion factor, V_S to the volume current and $GF_{S,E}$ to a predefined geometrical factor for the stack, V_C to the leak rate and $GF_{C,E}$ to the geometrical factor for the containment and p/p_C to the ratio of the environmental and the containment pressure.
It is interesting that in a first approximation the geometrical factors are largely independent on the spectral energy distribution of the gamma radiation and therefore also on the nuclide mixture. For this reason it is not necessary to have further informations on the activity of the discharged nuclide classes (noble gases, iodine, aerosoles) which cannot be gained easily anyway.
Additionally the assumptions and restrictions of the method are discussed.

1. Introduction

The assessment of the radiological situation in the plant and in the environment during an accident would be based on the measurement of high-dose-ionisation chambers installed in the stack and the containment (e.g. the so called "RABE"-monitors in the Swiss NPPs). In the first hours or days it is of special importance to estimate the immersion-dose rate in the vicinity of the NPP as quickly as possible using simple assumptions on the leakage path (ground or stack release). The calculation which presently would be applied is a two step method: first the release rate is estimated from the dose-rate measurements by assuming a nuclide mixture, then the immersion-dose rate is calculated basing on the same nuclide mixture and on the standard dispersion methods. Indeed, the assumption of a nuclide mixture is necessary for the determination of the release rate but - in good first approximation - not for the conversion of the dose rate in the stack or the containment to the immersion-dose rate in the environment.

To illustrate the last point we propose to compare the results of two identical dose-measuring devices in the environment and in the stack or the containment respectively. Doing this, we see that the signals of the two devices would differ only because of the different geometries and concentrations (e.g. due to the dispersion) at the points of measurement. Therefore, the dose-rate measured by the device in the environment may be written as a function of the dose-rate measured in the stack or the containment using eq. 1 or 2 respectively. Whereas

neither the pressures p and p_C nor the dispersion factor χ show any dependence on the energy and, therefore, on the nuclide mixture, the only parameter which introduces an energy dependence to these equations is the respective geometrical conversion factor GF. However, it can be demonstrated by model calculations that the energy dependence of the geometrical conversion factors can be neglected obtaining good first approximations of the dose rate in the vicinity of the NPP. The fundamental considerations for these model calculations and some results are shown in the following sections.

2. Basic considerations for the estimation of the energy dependence of the geometrical conversion factors

The dose-rate D measured by an ideal ionisation chamber, i.e. a ionisation chamber whose dose indication is independent of the energy, can be calculated principally for every geometry by the following simple formula:

$$D = DF_G \cdot C_G . \qquad \{eq. 3\}$$

Therein, C_G describes the concentration of the activity within the air [Bq/m³] of a geometry G. DF_G denotes the dose-rate-conversion factor which takes into account the geometrical distribution of the activity, the radiation absorption and the dose buildup in the air as well as the energy dependence due to other effects. Hence, for a monoenergetic source the dose-rate-conversion factor $DF_G(E)$ can be written as:

$$DF_G(E) = \gamma \cdot \int_G \frac{e^{-\mu r} \cdot \{1 + C \cdot \mu \cdot r \cdot e^{D\mu r}\}}{r^2} \cdot dV. \qquad \{eq. 4\}$$

The "Gamma" constant γ defines the gamma-dose rate of a point source per unit activity in a distance of 1 m from the source. μ denotes the linear attenuation coefficient and r the distance of the ionisation chamber from a volume element dV. The expression in the curved brackets (Berger's formula) describes the energy deposition buildup factor; the constants C und D used therein are determined empirically. The parameters γ, μ, C und D depend explicitely on the energy E of the Gamma-radiation. The integration has to be performed over the entire space of interest, e.g. the volume of the stack, the containment or the semi-space of the environment.

For nuclide mixtures having multiple gamma emission lines one obtains the dose-rate-conversion factor for a defined geometry as a weighted sum over all monoenergetic dose-conversion factors:

$$DF_G = \sum_{i,j} p_i \cdot DF_G(E_{i,j}), \qquad \{ eq. 5\}$$

where i denotes the nuclide, j the emission lines of the nuclides. p_i describes the contribution of the nuclide i to the total activity of the mixture.

The monoenergetic conversion factor $GF_{G1,G2}(E)$ now can be definded as

$$GF_{G1,G2}(E) = \frac{DF_{G2}(E)}{DF_{G1}(E)} . \qquad \{eq. 6\}$$

For our case of the NPP, G1 is the geometry of the stack or the containment, G2 that of the environment.

Finally, the total (mixture dependent) geometrical factor $GF_{G1,G2}$ for the conversion of the dose rate from the geometry G1 to G2 is defined as the ratio of the two dose-rate-conversion factors:

$$GF^{tot}_{G1,G2} = \frac{DF_{G2}}{DF_{G1}} = \frac{\sum_{i,j} p_i \cdot DF_{G1}(E_{ij}) \cdot GF_{G1,G2}(E_{ij})}{\sum_{i,j} p_i \cdot DF_{G1}(E_{ij})} \quad \text{(eq. 7)}$$

Hence, the total geometrical conversion factor can be written as a weighted average of the monoenergetic geometrical conversion factors.

3. Results and discussion

From equation 7, it is obvious that the total geometrical configuration factor for a nuclide mixture has to be smaller (higher) than the highest (lowest) monoenergetic configuration factor.

The range of the monoenergetic geometrical conversion factors for spherical and cylindrical geometries to the semi-infinite volume of the environment are presented in the following table. The cylindrical geometry is an approximation for the stack: the height (75m) and the radius (3m) as well as the position of the measuring ionisation chamber at the center-bottom of the stach were chosen to conform with one of the Swiss NPP's. Besides, we assumed that the containment may be approximated by a sphere with a radius of 8m with the ionisation-chamber in the center of the sphere. The other spherical geometries have been chosen to show the effect of the volume.

Table: Energy-dependent geometrical conversion factors and proposed "best"value

Energy [keV]	$GF_{S,E}$: Cylindrical geometry (Stack) to half space; height=75m, radius=3 m, dose at bottom of stack	$GF_{C,E}$: Spherical geometry (Containment) to half space			
		radius=3m, dose at center of the sphere	radius=8m, dose at center of the sphere	radius=20m, dose at center of the sphere	radius=50m, dose at center of the sphere
50	34.2	28.3	9.7	3.4	1.2
100	56.1	46.0	16.1	5.7	2.0
200	54.1	42.9	15.7	6.0	2.2
500	55.4	43.6	16.1	6.3	2.4
1000	59.7	46.3	17.3	6.9	2.8
2000	70.3	54.3	20.4	8.2	3.4
4000	86.1	67.5	25.3	10.2	4.2
Proposed "best" value	60	45	17	6.5	2.5
Variations (50-4000 keV)	±50 %	+50/-40 %	+50/-40 %	+60/-50 %	+70/-50%

From our considerations in chapter 2 and in view of the numerical calculations the direct conversion of the dose measured in the stack or the containment to the immersion dose in the vicinity of a NPP by the means of averaged geometrical conversion factors (called proposed "best" value in the table above) seems to be a good method to obtain reasonable first estimates. For the simple geometries we used for the calculations the errors which are introduced neglecting the dependence of the geometrical factor on the energy or on the nuclide mixture respectively are distinctly smaller than a factor of 2. This error has to be judged with respect to other errors which may be much more important, stemming for example from meteorological or topographic uncertainties. However, if a better accuracy has to be attained, it is possible to determine a geometrical conversion factor which depends on the time between the start of an accident and the release of the radioactivity to the environment, since the time-dependent composition of the inventory of the core of an NPP is known quite well. Nevertheless, the additional effort to perform these calculations seems to be excessive for most of the needs.

The determination of the geometrical conversion factor $GF_{C,E}$ taking into account the shielding and scattering effect of the installations and (concrete) walls within a real containment of a NPP is usually more costly than our simple calculations above and has to be performed using numerical integration methods. The geometry of a stack on the other hand is usually approximated quite well by a cylindrical shape similar to that above so that the calculation of the related geometrical factor $GF_{S,E}$ can be performed in a much simpler way than for the containment. However, these calculations have to be done once as a precautionary action in order to allow fast, simple estimations of the immersion dose in the environment using the measurements in the containment and the stack.

Conclusions

- The immersion-dose rate (dominated by noble gases) at the "most-affected" point in the vicinity of a NPP can be determined from the dose rate measured in the stack or the containment respectively using the formulas presented in the summary.

- For good first estimations of the immersion-dose rate, it is reasonable to use an averaged "best" value of the geometrical conversion factors GF. For a cylindrical or a spherical geometry approximating the stack or the containment, this leads - independently of the emitted nuclide mixture - to a maximum error which is less than 50%. The errors introduced by other parameters, e.g. the dispersion coefficient χ for the actual weather conditions, are usually much more important than this.

- The geometrical conversion factors can be determined once for the actual stack and containment of a NPP as a precautional measure. In case of an accident, this allows very fast estimations of the immersion-dose rate in the vicinity of the NPP which faciliates the timely decision upon the necessary steps to be taken.
  The estimation of the immersion dose is of special importance, because it yields the main contribution to the dose to the population just after an accident.

Indeed, it is possible to determine a nuclide-mixture-dependent geometrical conversion factor for mor accurate calculations performing a weighted average of the energy-dependent factors. However, this is paid with the abandonment of the main advantage of the method, i.e. the possibility of very fast estimations of the environmental immersion-dose rate.

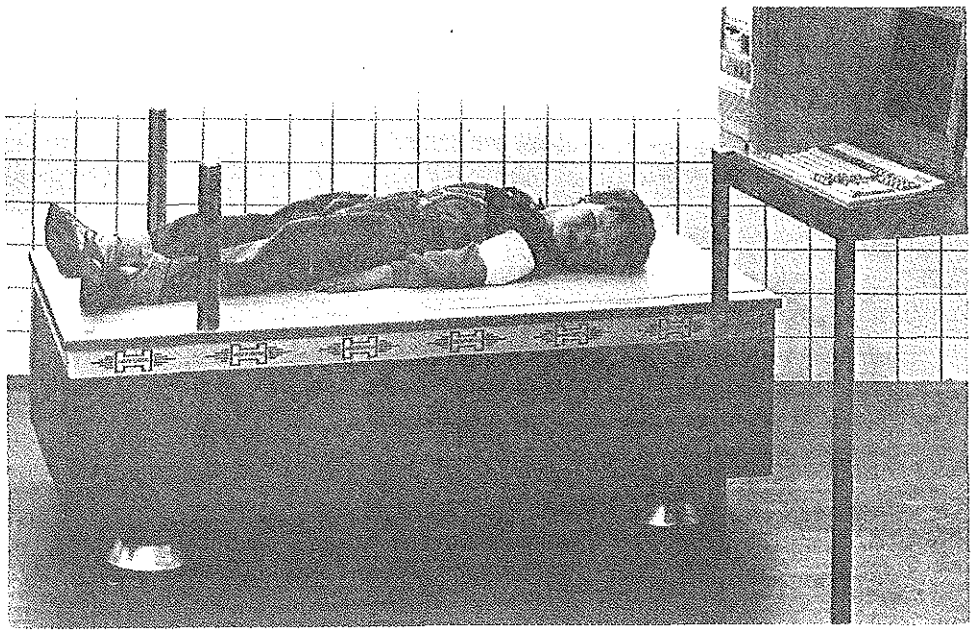
INTERNAL CONTAMINATION MONITOR FOR QUICK SERIES EXAMINATIONS

M. Nöckel
Herfurth GmbH, Beerenweg 6 - 8
2000 Hamburg 50

The internal contamination monitor H 13010 is designed for the monitoring of persons with possible ^{137}Cs contamination in the muscular tissue, in the digestive tract and in the lung.

In general the considerations provide for measurement of the torso and the back muscular tissue including buttocks, so that the person to be measured is measured best lying on the back.

Therefore, a measuring table has been developed to position the person to be measured on the back. This position makes it possible to measure an adult from the shoulders to the thighs with one large-area gamma plastic detector (80 cm * 50 cm).



Measuring table

To increase the detection limit of the monitor, the detector is surrounded with a tub-shaped lead shielding of 50 mm thick lead stones. This way the influence of ambient radiation is reduced by a factor 7.

As the dominating leading isotope ^{137}Cs is known, a nuclide specific measurement, requiring relatively long measuring times, is not necessary.

With the H 13010 an integral measurement is carried out, which achieves higher efficiency by use of the Compton effect, thus

permits shorter measuring times in the range of seconds.

Due to the lead shielding the detector supplies a background pulse rate of 550 cps, if the ambient radiation is 0,1 µSv/h. The efficiency of the detector for ^{137}Cs is about 6 % if the point-source is shielded by a 5 cm thick water layer (corresponding to the absorbtion of the body tissue) - factor about 0,5 for 662 keV.

According to DIN 25482 part 1 (approximation formula), a recognition limit of 120 Bq and a detection limit of about 240 Bq results from a measuring time of $t_o = t_b = 60$ s and the quantiles $k_1-\alpha = k_1-\beta = 1,65$ (corresponds to an error of 5 %).

The H68k-computer acquires the detector pulses and processes them for display of the activity. Furthermore, during measurements, calibrations and in case of service, the user is guided by a dialogue with the terminal displayed on the video.

```
+-------------------------------------------------------------------------
| ==== Measuring Mode ================ Herfurth H13010 ===
|
|              >>>>>>>> READY TO MEASURE <<<<<<<<
|
|  Personal - ID     : x xxxxxxxxx     ( 1st character:
|                                      F= female    over 18 years
|        - height: zzz cm (zzz cm)      M= male      over 18 years
|                                      T= teenager  12 - 18 years
|        - weight:  zzz kg (zzz kg)     C= child     3 - 12 years
|                                      B= Baby      0 -  3 years
|        - Comment: _____________________________________________
|  Inputs correct?
|     Then position person and start measurement with key F1.
|
-------------------------------------------------------------------------
```

Besides of filing about 4000 measurements and selecting these through a computer interface available, measuring records can be printed out through the printer interface after each body measurement.

Personal-ID	Height cm	Weight kg	Activity kBq +/-kBq	Dose mSv +/-mSv	Date	Time

The components of the internal contamination monitor only consist of an electronics housing, a terminal and the measuring table, which, due to its low centre of gravity can easily be built-in into vehicles, without any additional supports. The total weight of the monitor comes to about 1000 kg.

Calculation of the Detection Limit

The desired detection limit of the internal contamination monitor H 13010 is only depending on the selected measuring times, the duration of the background measurement and the measurement of persons (parameters), if the background pulse rate at the location of setup is available.

Acc. to DIN 25482 part 1 "Nachweisgrenze und Erkennungsgrenze bei Kernstrahlungsmessungen" (Detection limit and recognition limit of nuclear radiation measurements) The required measuring time for persons can easily be defined by using the approximation formula, if the duration of the background measurement is known.

By converting the approximation formula to the measuring time for persons t_b, there will be following formula:

$$t_b = \frac{\varrho_o * t_o}{t_o \left[\frac{\varrho_n^*}{(K_1 - \alpha) + (K_1 - \beta)}\right]^2 - \varrho_o}$$

with: t_o = Background measuring time in s

$k_1-\alpha$ = 1,65 for error probability α (5%)

$k_1-\beta$ = 1,65 for error probability β (5%)

ϱ_n^* = expected net count rate

ϱ_n^* = expected background count rate

The expected background count rate ϱ_o multiplied by the background measuring time t_o results in the pulse sum to be attained from the background parameter. This means that set parameter figure value of the pulse sum to be attained, can be put in for the numerator of the formula.

The background measurement time t_o is calculated from the "pulse sum to be attained" divided by the background count rate of the channel status display.

The desired detection limit DL x detector efficiency n, can easily be defined with help of the following formula and with reference to the desired detection limit the necessary measuring time for persons t_b, if it is used instead of the expected net count rate ϱ.

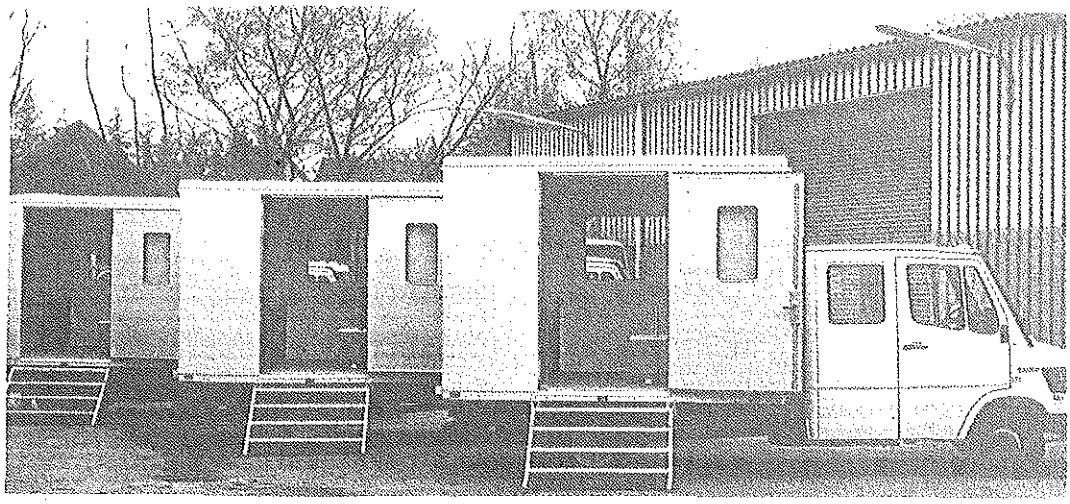
$$t_b = \frac{pulse\sum}{t_o \left[\frac{NWG * \eta}{3,3}\right]^2 - \varrho_o}$$

The activity value stated at the end of measurement of a person is calculated in the system by subtracting the background pulse rate from the gross pulse rate after the measurement is finished. and by division with the detector efficiency.

The absolute error of the activity, which is also supplied, consists of the errors from the background measurement and from the calibration measurement.

The alarm threshold, to be set as parameter, can be put in directly in the Bq value of the activity

If after a body measuring the measured activity value exceeds this alarm threshold, the measuring result is marked with "alarm level exceeded".



Vehicles ready for operation

Background Shielding by the Person to be Measured

Based on the lying arrangement of the measurement, the background of a measurement is reduced by the person to be measured. The person to be measured has the effect of an additional detector shielding. Investigations at the Paul Scherrer Institute have given an approximation for the individual body shielding in a linear context to the body weight of the person. If the body weight of the person is entered, the reduced background value is defined in linear formula y = mx + b by the computer system and is deducted from the measured gross pulse rate.

In general it may be said that for an average middle European man with average body size (75 kg) and a natural ambient radiation of 0,1 µSv/h the background value of the measuring table is reduced by about 10 %.

Calibration Factor depending on Body Weight

The phantom calibrations at the Paul Scherrer Institute also resulted in different calibration factors depending on the simulated body weight for the measuring arrangement. An analysis of the calibration results showed an approximation of an e-function for the formula $y = x\ e^{-ug}$. According to this e-function now the individual calibration factor is calculated in the computer system, when the body weight of the person to be measured is put in.

The measuring equipment supplies a sensitivity of about 4,5 % for ^{7}Cs at a body weight of 75 kg

Data Filing System EVI

Acquisition of the measuring data is done by reading the measuring value memory of the appertaining internal contamination monitor H 13010.

The read measuring data are filed in a data base table and can be demonstrated on the display respectively recorded on the printer with various functions.

In an administration table the EVI-system logs the daily quantity of measurements for the days on which measurements have been made. Additionally the time period of the measurements as well as the filing remarks are entered.

Measurements can be stored on discs by days with the help of the filing function. Measuring data on discs can also be displayed or recorded.

MOBILE WHOLE-BODY COUNTER FOR BABIES AND TODDLERS

P.Hill
Forschungszentrum Juelich Gmbh,ASS, Postfach 1913, D-5170 Juelich

Summary

Construction and calibration of a whole-body counter especially designed for monitoring body burdens of babies and toddlers are described. Cs-137 incorporations of as small as 150 Bq can be detected in 1 minute of measuring time by 1-2 NaI(Tl)- detectors contained in a radiation reduced steel shielding. The counter is suitable for use in mobile laboratories operating in areas (e.g. CIS) contaminated by fall-out from nuclear accidents.

1. Introduction

A special whole-body counter for children has been developed for use in mobile labs. Dose conversion factors for small children known to be up to ten times higher than for adults it is important to obtain their body burden more reliably than it is possible with most of the body counters designed for adults.

2. Construction

A bed-type configuration has been chosen for the counter (fig.1). One or two of Bicron's large NaI(Tl)- detectors (7.6 cm x 12.7 cm x 40 cm) can be put into a radiation reduced steel shielding of 5 cm thickness. Children are to be placed in a height adjustable half- tube . The counter is put onto four wheels and the steel plates can be easily removed. Thus the system becomes highly transportable and flexible in operation. Though the system is designed preferably for use in mobile laboratories it may easily been taken out and used e.g. in hospitals or child wards.

Using a measuring time of only 60 seconds limits of detection of 150 Bq Cs-137 (1 detector) and 100 Bq Cs-137 (2 detectors) can be achieved.

The data acquisition is PC-based. The ACCUSPEC MCA card together with ABACOS software (Canberra) is used for taking data and evaluating body burdens.

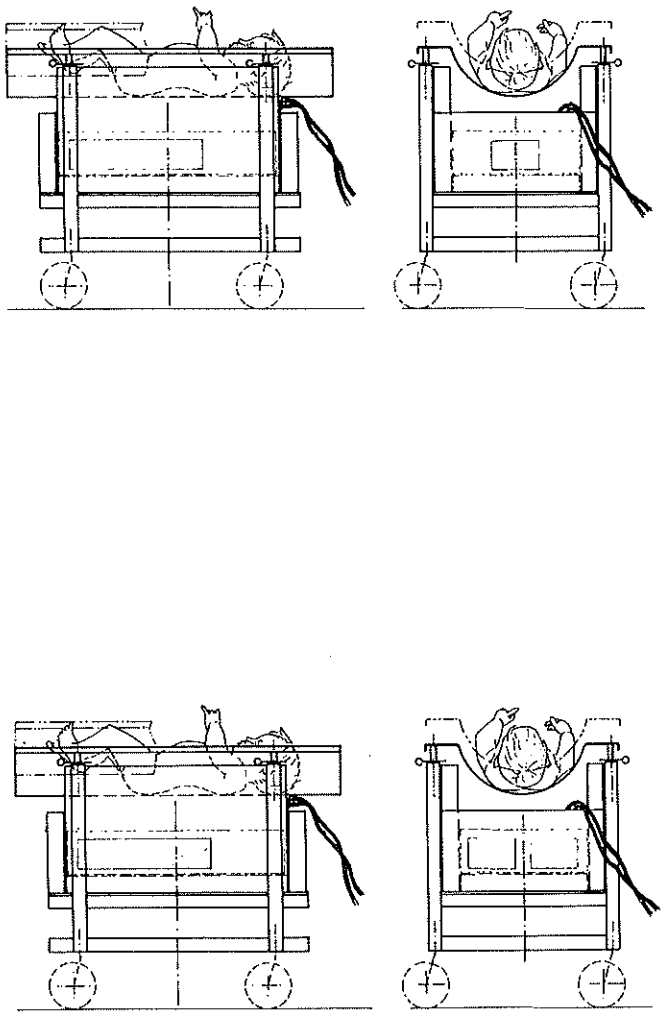


Fig. 1 : Drawing of the KFA whole-body counter for small children showing the position of the detectors in the shielding .
Upper part : 1 detector arrangement
Lower part: 2 detector arrangement

3. Calibration

The KFA bottle phantom has been used for calibration. Bottle volumes of 0.25, 0.5 and 1 l allow to simulate the body shapes of children of 4 kg, 8 kg, 12 kg and 14 kg weight.

The 4 kg phantom consists e.g. of a 1 l bottle representing th head, one 1 l and one 0.5 l bottle represent tthe body. Two 0.25 l bottles simulate hands and arms. Four 0.25 l bottles are put in place of the legs.

A solution of Cs-137 in water has been used for calibration. In the lowest position of the adjustable half-tube sensitivities of 1.8 cps/100Bq (4 kg) , 1.2cps/100Bq (8 kg), 0.98cps/100Bq (12 kg) and 0.93 cps/100Bq (14 kg) for Cs-137 have been determined.

The relative efficiencies for different positions of the half tube have been determined in the case of the 4kg phantom to be 1:0.71:0.52 for heights of 0 cm, 5.5 cm and 11 cm . Heights are given relative to the lowest position.

4. Operation

Three counters are used 1992 by a German Chernobyl burden measuring programme in Russia, Byelorussia and Ukraine. One of them operates independently, whereas two of the counters operate in connection with FASTCAN- systems (Canberra) sharing the data acquisition systems. By appropriately adjusting the height of the half-tube where the children are positioned, the same efficiency curve can be applied for both systems. Correction factors account for the weight dependence of the efficiency curve.

A Cs-137/Co-60 source is used for a daily calibration check. Also environmental background is checked daily. Though it mainly arises from surface contaminations the use of water phantoms is recommended in determing background to account for the reduction of direct background radiation (e.g. from Cs in trees) by the measured children.

The bed-type configuration allows the mother to maintain eye contact to the child and even to touch it during the measurement, which we feel to be important for the wellness of the child and keeping it quiet. The influence of a possible body burden of the mother has been simulated with a 37 kBq Cs-137 source and found to be neglegible at arm's length distance to the child.

Acknowledgements

Funds have been provided by the German Ministry of Environment (BMU). The help of Ch.Kuehn in data taking is greatfully acknowledged.

RESULTS OF A DOSIMETRY PROJECT IN THE UKRAINE

Christian Wernli and Markus Boschung

Paul Scherrer Institute (PSI), Villigen, Switzerland

Summary

n Summer 1991 a team of Swiss and Ukrainian specialists performed about 3400 whole body and 000 foodstuff measurements in the vicinity of Polesskoje, 50 km west of Chernobyl. For this roject, led by the Swiss Disaster Relief Unit, a mobile laboratory with whole body counter and NaI gamma spectrometer for food and environmental samples was built under guidance of the Radiation Metrology Section at PSI. The results show that in the 27 villages visited the mean nnual internal dose from Cs-137 was in 1991 about 0.16 mSv/a with a maximum of 9 mSv/a. It as been confirmed that the highest activity concentrations in environmental samples, including oodstuffs, occur in products from the wood. The project is continued in 1992.

ntroduction

n reply to a call for help of the Ukrainian Government a project of humanitarian aid was set up by he Swiss Disaster Relief Unit (SDR). This organisation is designed to intervene in case of natural r manmade disasters, war, famine, and assistance to refugees. The SDR is part of the Directorate Development and Humanitarian Aid in the Federal Departement of Foreign Affairs.

he objectives of the project were:

- to support and assist the Ukrainian authority in measuring whole body activity and activity concentrations in food.
- to inform directly the concerned population about radiation levels.
- to educate and reduce fear of the population.
- to aquire experience in case of further accidents.

o achieve this task the Radiation Metrology Section at PSI was asked for technical advice and upport. Based on their proposal it was decided to build a mobile radioactivity laboratory with a hole body counter, a food counter, and several dose rate and contamination measuring devices.

echnical Information of the Mobile Laboratory

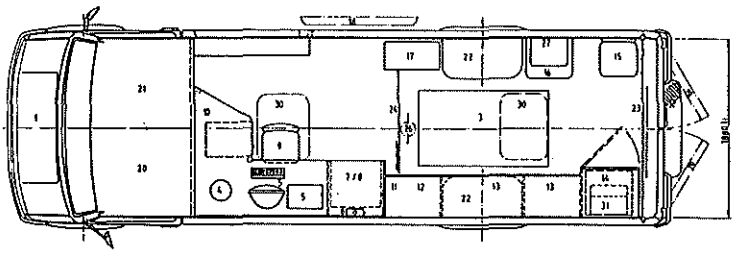


Figure 1: *Schematic view of the mobile laboratory with the whole body counter (3) and the food counter (4)*

The laboratory was built in a 7 ton Mercedes van (model 711) with side door for access to the food counter and entry from the back side for whole body measurements (figure 1 and 2).

Figur 2: *View of the Mercedes van*

Technical details of the whole body counter:

Manufacturer:	Herfurth
Detector material:	Plastic scintillator
Detector dim.:	80 x 50 x 10 cm
Shielding:	Tub of lead, 5 cm wall thickness
Background:	400 cps at 0.1 µSv/h
Measuring time:	60 sec
Geometry:	Lying on the back
Efficiency:	6 % whole body Cs-137
Detection limit:	1 kBq Cs-137

Technical details of the food counter:

Detector material:	NaI
Detector dim.:	3 " x 3 "
Shielding:	Lead castle, 5 cm wall thickness
Electronic:	PCAP-Board (HV, ampl., ADC)
Measuring time:	5 min (15 min)
Geometry:	1 L-flask (1 L-marinelli)
Resolution:	8 % at 662 keV
Detection limit:	50 Bq/L (20 Bq/L) Cs-137

Summary of the Project

Based on a letter of understanding between the Ministry of Health of the Ukraine and the Swis Disaster Relief Unit, signed in January 1991, a mission in the district of Polesskoje was planned Polesskoje is located 50 km west of Chernobyl. The surface contamination by Cs-137 in this regio is extremely inhomogeneous and varies between less than 30 and about 2200 kBq/m^2. In 1991 Cs 134 was only about 10 % of Cs-137. The external dose rates vary between 0.1 and 10 µSv/h. In thi district of 1300 km^2 area located immediately west of the 30 km zone 27'000 inhabitants lived i 1990. In 1991 15 out of a total of 63 villages were evacuated and for another 16 villages evacuatio is considered.

The central station of the mobile laboratory was at the hospital in Polesskoje. From here, a team o 2 Swiss and 4 Ukrainian specialists, including a medical doctor, went out to the villages to measur whole body activities and foodstuffs. In the time span of July to October 1991 27 villages wer visited and 3440 whole body and 960 food measurements were performed. In total 7 Swis dosimetry specialists joined the project for one month each. At the same time a Swiss medical tean was involved in a medical aid project of SDR at the hospital of Polesskoje. At the end of bot projects a sociological survey was made in a Ukranian / Swiss cooperation. This was to find ou how the population responded to the two aid projects.

Results

The local inhomogeneties of average whole body activities are shown in figure 3.

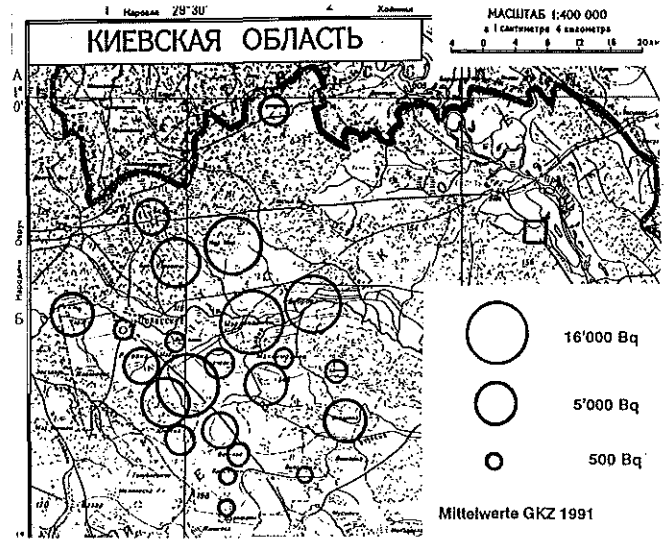


Figure 3: *Average whole body activity (Cs-137) per village. The areas of the circles are proportional to the mean of the measured activities. The square on the right is at the location of the nuclear power plant.*

The age distribution of the measured population is given in figure 4. Since the project was based on humanitarian aid and not on a scientific program, the persons to be measured were not selected. It can be seen from figure 4 that mainly children and older persons showed up. Most of the working population did not find time to go to the counter. The weight distribution in figure 5 shows a first peak due to the large number of children. It can also be seen that a few persons are remarkably heavy. The distribution of the whole body activities is given in figure 6. The highest value was 380 kBq.

For dosimetric calculations it was assumed that the whole body activity is constant over the whole year. For each person the specific activity in Bq/kg was determined. Then a dose factor of 2 µSv/a per Bq/kg was used to calculate individual dose. The mean and maximum dose values for different age groups are given in figure 7. There is a slight increase in dose with age. It is assumed that this is due to the fact that imported non contaminated food is given mainly to the children. The overall average dose from incorporated Cs-137 was 0.16 mSv/a in 1991. The highest value was 9 mSv/a.

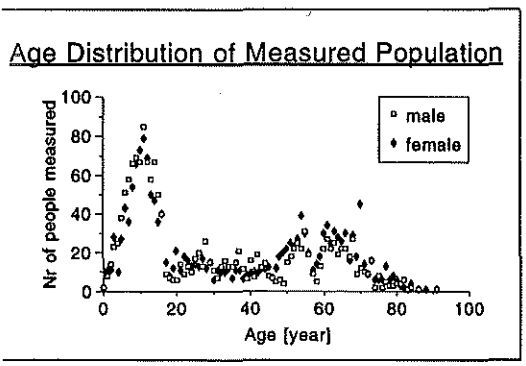


Figure 4: *Age distribution of the measured population*

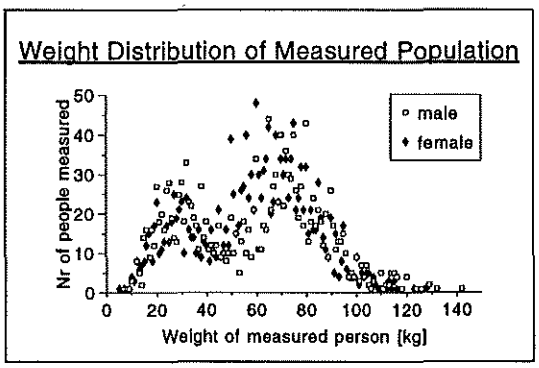
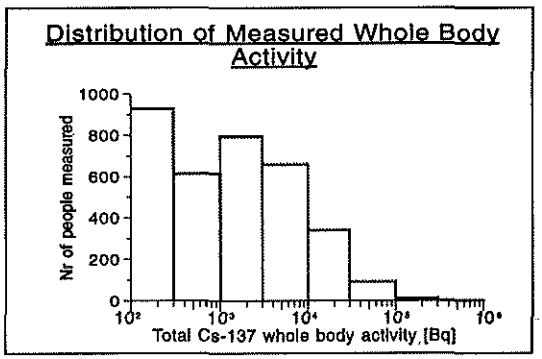


Figure 5: *Weight distribution of the measured population*



Figur 6: *Distribution of the whole body activities of the measured population*

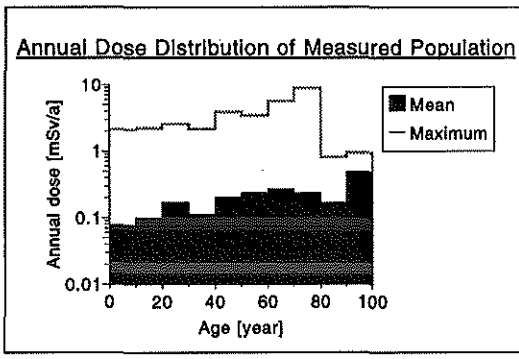


Figure 7: *Annual dose distribution of the measured population with mean and maximum dose values for different age groups*

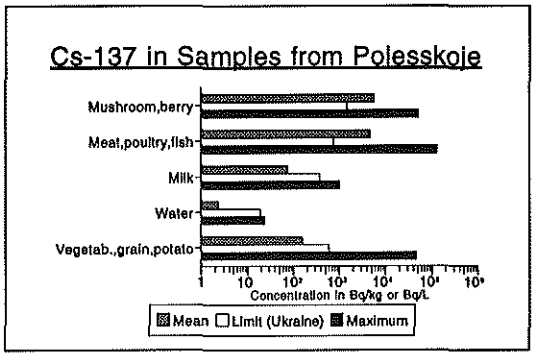


Figure 8: *Mean and maximum values for activity concentrations in groups of foodstuffs compared to the Ukrainian limits*

For a small number of persons measured fo Cs-137 urine samples were taken and a Sr-9 and an alpha analysis were made at PSI. A expected from measurements on the groun contamination Sr-90 was found to be on much lower level than Cs-137 and no alph emitter could be detected.

In some villages milk and eventually othe food is banned. The foodstuff measured wa mainly locally produced. In figure 8 the mea and maximum values for activities in group of foodstuffs are given and compared to th Ukrainian limits. It has been confirmed tha mainly products of the wood (berrie mushrooms, wild animals) contain the highe Cs-137 contaminations.

Conclusion

The measurements agreed quite well with previous results of the Ukrainian authorities. Thi confirmation of the actual contents of radioactivity in whole body and food by an independen foreign team of experts was welcomed by the population. The study shows that there is a wid variation in resulting dose. Higher doses often occur if the advice of the Ukrainian authorities t ban specified foodstuffs is not followed.

The sociological study shows that the population is interested in independent measurements an interpretations of the radioactivity. The new mission in 1992 indicates that to day the socic economic problems increasingly dominate over the fear of radiation.

EXPERIENCE WITH MOBILE NaI(Tl)-GAMMASPECTROSCOPY FOR THE FAST ASSESSMENT OF RADIONUCLIDE CONTAMINATIONS IN THE OPEN AIR

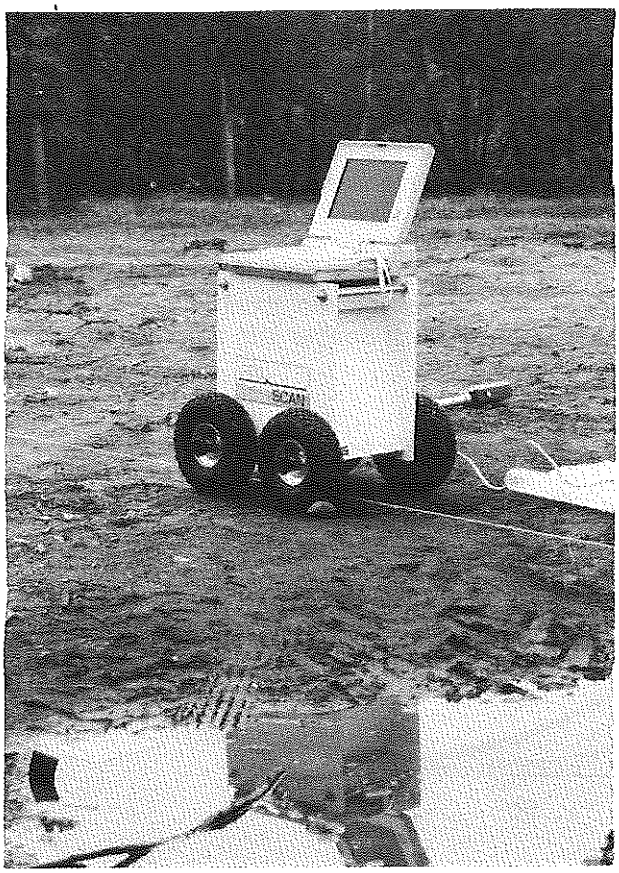
Volker Genrich
Genitron Instruments GmbH, Heerstr. 149, D-6000 Frankfurt/Main 90

Summary

The author will report on the application of a mobile gammaspectroscopic set-up, which was used for scanning a Caesium-137-contamination within the terrain of an radioisotope production plant. Radionuclide concentrations in the soil have been determined simultaneously for the artificial nuclides Cobalt-60 and Caesium-137, and for the natural nuclides Potassium-40, Radium-226 and Thorium-232.

1. The basic components of the mobile gammaspectroscopic set-up

The spectroscopic equipment is based on a PC-laptop and a MCA-card with 1024 channels. Detector and preamplifier are mounted inside a collimator, providing a viewing angle vertically down +/- 45 °. All components are mounted on a special trolley, called *"GeoSCAN"*. Equipped with four pneumatic tyres, the whole equipment is suitable for cross-country work (see fig. 1).



Using a 3" x 3" NaI(Tl)-crystal, measuring time is a few seconds per data point. A grid with a spacing of 1m x 1m is picked up within a very short time. By activating different software tools on the PC, the in-situ-spectra can be evaluated directly in the field. Finally, five different nuclide concentrations are available. Even on the trolley, contour maps (isolines) or 3D-plots of the contaminated area can be examined.

Figure 1:
Routine work at 5 ° C in the open air with the "GeoSCAN". The equipment did not show any complication (even in hostile environment)

- ❑ exact radionuclide mapping around uranium mills, on tailings etc.
- ❑ scanning on sites, previously contaminated by industrial use of radionuclides
- ❑ clearing outdoor ground after decommissioning (NPPs, accelerators etc.)
- ❑ fast assessment of accidental releases into the soil (transportation accidents)
- ❑ in-situ tracing for radionuclide effluents, deposed in beach and river sediments
- ❑ airborne or carborne profiling in emergency situations

Table 1 : *Fields of application for mobile NaI(Tl)-Gammaspectroscopy*

2. What is the difference to beta-contamination monitoring ?

Without any doubt, the equipment for gammaspectroscopic scanning is much more sophisticated (and expensive) than a standard contamination monitor (equipped with a proportional counter). Why then look for gamma radiation with "heavy" equipment?

The difference is the "third diemension"! Although proportional detectors exhibit a certain gamma sensitivity, in practice their gamma-efficiencies are too poor (less than 1%) for fast and effective measurement. Furthermore, measuring only beta radiation, such equipment can only detect **surface contamination** in Bq/m^2: any activity, that has penetrated farther than 1 mm into the ground, cannot be discovered.

Consequently, under open-air conditions a beta contamination monitor can only detect a small portion of the total activity potential. For checking the environment, it is a much better approach to use gamma radiation monitoring and assess Bq/kg.

As we can gather from literature, in-situ detectors mostly have been used without a collimator. In this case, an area of more 10 m in diameter and of less than 20 cm in depth will contribute to the detector signal. As our mathematical modelling shows (figure 2), a collimated detector is able to look much deeper into the soil (by excluding the "thin" layers far outside the center area).

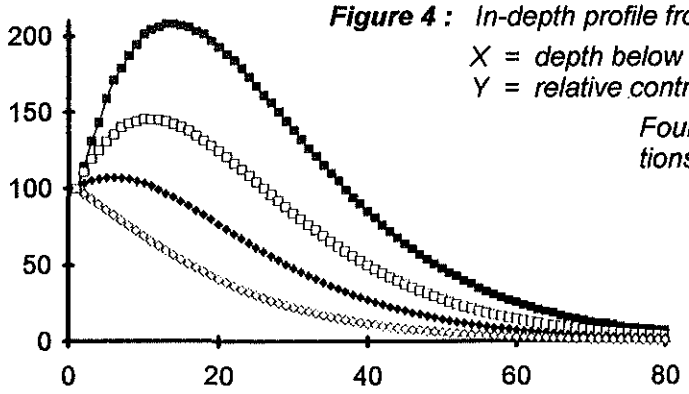


Figure 4 : *In-depth profile from a collimated detector:*
X = depth below ground surface (cm)
Y = relative contribution to the signal (%)
Four different detector elevations above ground surface:
$H_{(eff)}$ = 5 cm
$H_{(eff)}$ = 10 cm
$H_{(eff)}$ = 15 cm
$H_{(eff)}$ = 30 cm

3. Typical applications :

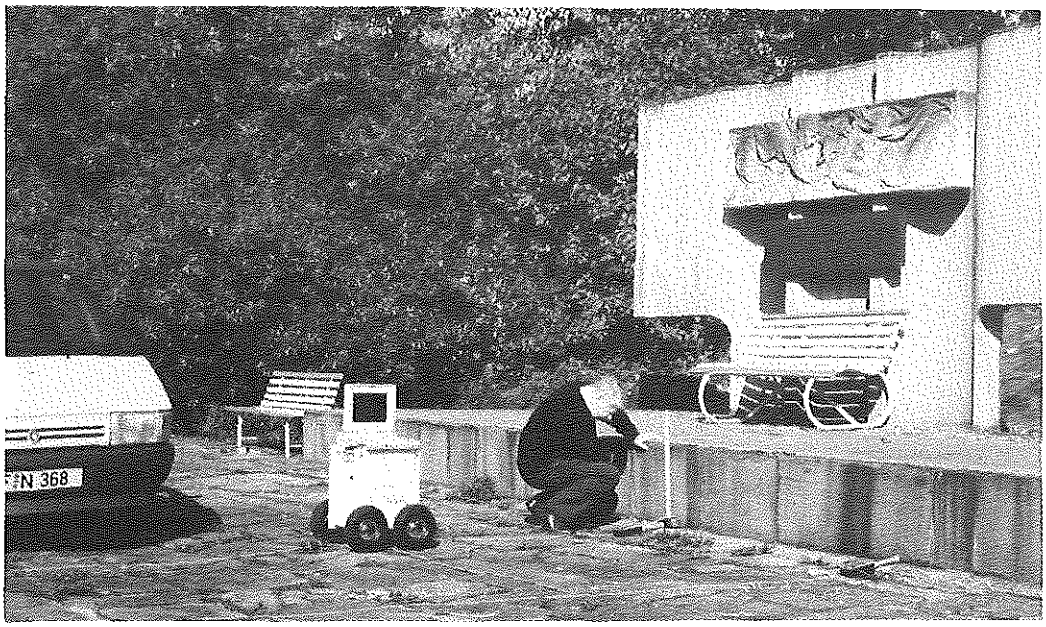
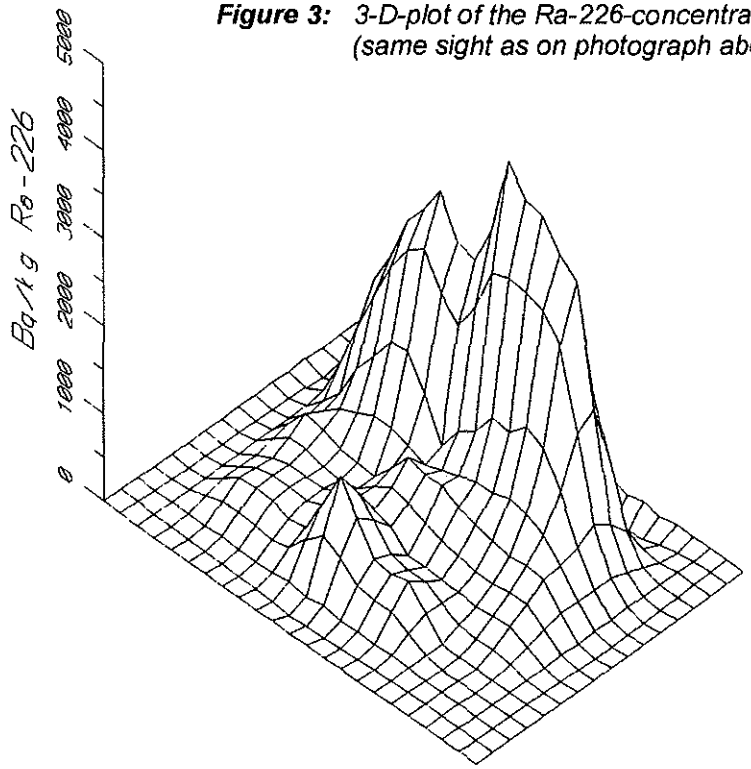


Figure 2: *The monument of Mm. Curie, built on the ruins of the "historical" uranium-color factory at Jachymov/CSFR. Ground contaminatin: 5000 Bq/kg Ra-226*

Figure 3: *3-D-plot of the Ra-226-concentration (same sight as on photograph above)*



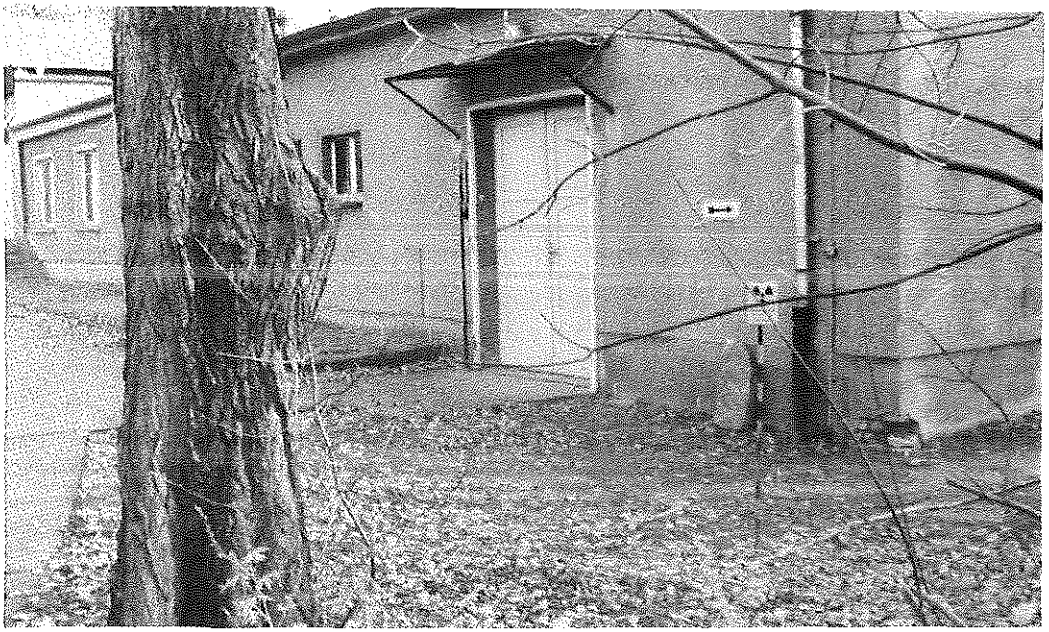
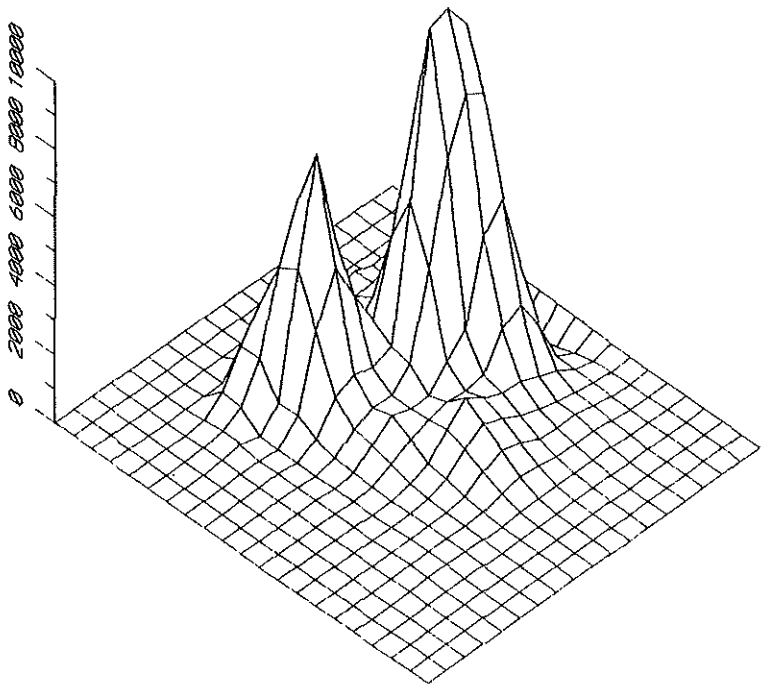


Figure 4: *Area in front of a building, thas has been used for waste-water conditioning from a radioisotope production plant.*
Ground contamination: 10 000 Bq/kg Cs-137

Figure 5: *3-D-plot of the Cs-137-concentration (same sight as on photograph above)*



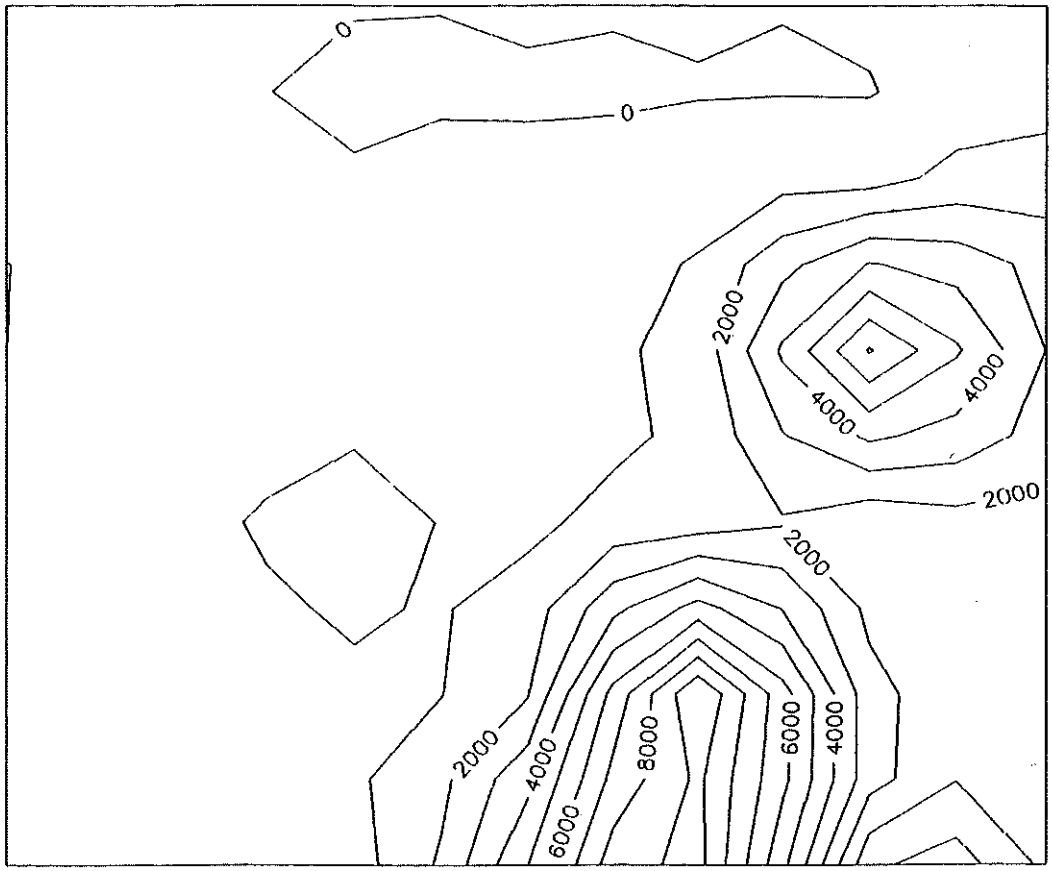
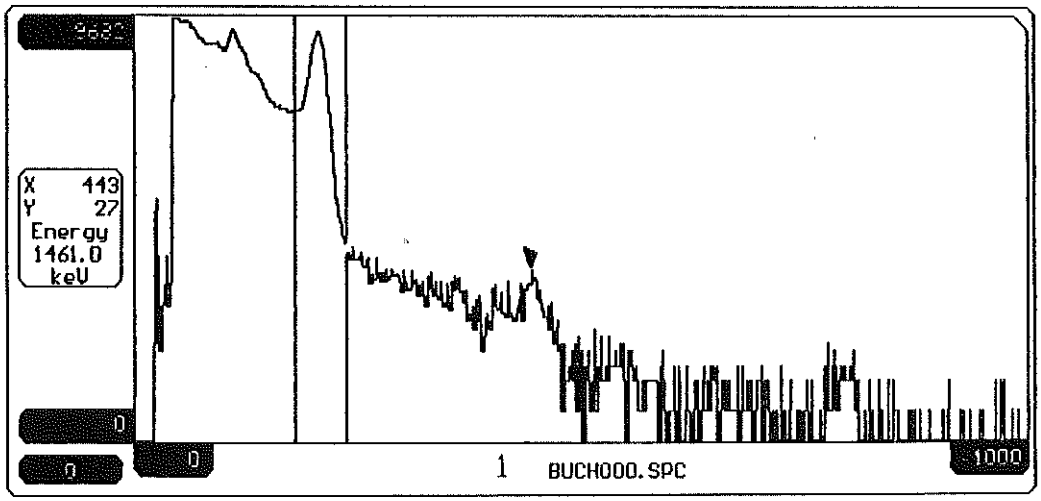


Figure 6: *By using the contour map, the two contaminated spots can be removed very precisely. Thus, a minimum of caesium-contaminated soil has to be dumped .*

Figure 7: *In-situ spectrum (log-scale) at the maximum of the contamination*



COMPUTER AIDED INSTRUCTION ON RADIATION PROTECTION

M. Mauersberger, K. Pingel
University of Siegen, Physics Departement, 59 Siegen, Germany

Summary

A computerprogramm using menu guided user interaction was developed to give a basic introduction to the topics and quantities related to radiation fields and radiation doses.

1.) Introduction.

A wellknown difficulty in instructions of radiation protection for people not yet familiar with this kind of danger is their difficulty to develop a "feeling" for its relevance. The activity of a source of radiation and the corresponding radiation doses are not easily connected but depend on special conditions. The personal risk connected with a given dose must be related to other risks experienced in life before an appropriate personal "emotional" scale is established necessary for an intuitive and adequate treating of every day laboratory situations with ionizing radiation.
The necessary experience is usualy not easily acheaved by the usual way used by books an lectures. We use a portable computer running a programm with a menu driven simulation of selectable situations in radiation protection.

2.) Instruction in radiation protection by a computer.

The program "RP-TUTOR", a prototype of which is presented, can the simulate variety of situations with radiation protectional relevance and the quick and easy change of parameters like: the kind of radiation, its energy, external or internal exposure, different shieldings, exposure time and geometry, the determination of radiation doses, the evaluation of a dose/risk relation, comparison with other risks, and other possible factors. Various sources of radiation can be selected or "created". Different contributions to the radiation and the accociated risks can be individually studied.
Fig. 1 shows a set of menus selected in sequence. Diagrams to show the dose/shielding or risk/dose relation can be displayed for the situation selected. In the present version of the program point sources and other simplifications are used, but an extension to more realistic cases is under way.
One further advantage of the simulation is also that the use of real radioactive sources can be avoided. Nevertheless in some instances experiments should support the instructions. Application of the program to design radioprotectionaly save laboratory experiments is supported.
The program relies on a course in radiation protection written by one of the authors [1].

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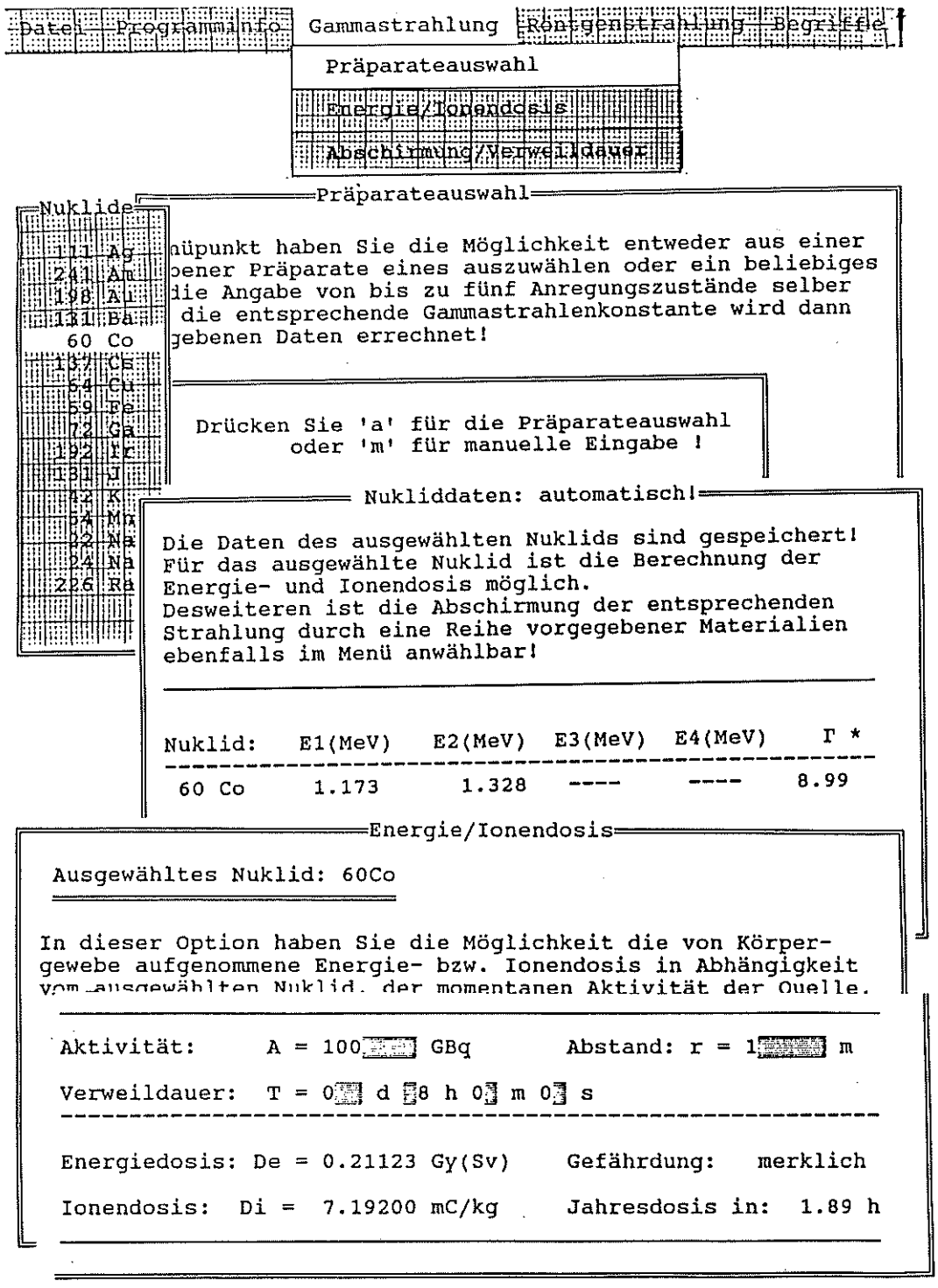
Fig. 1: A menu selection in the CAI - Program "RP-TUTORR".



AGRICULTURE, ENVIRONMENT AND NUCLEAR ACTIVITIES: HOW TO REACT IN CASE OF AN ACCIDENT

G. Griperay* and R. Coulon**

* Fédération Nationale des Syndicats d'Exploitants Agricoles, Paris, France
** Institut de Protection et de Sûreté Nucléaire, Fontenay-aux-ROSES, France

Summary

Presentation of a reference handbook analyzing the consequences of a nuclear accident on agriculture and how to react in order to mitigate them.

Based on the right of citizens to be informed on civil security organization, on the protection of population, goods and environment : this book is a preliminary reflexion to post-accidental planification.

1. Introduction

In the present world, communication is of major importance. The public asks for clear and complete information in order to understand and judge the basis of the options chosen by the political authorities.

In this context, a document entitled "Agriculture, Environment and Nuclear Activities : how to react in case of an accident (Agriculture, environnement et nucléaire : comment réagir en cas d'accident) was jointly prepared by a professional organization, the "Fédération Nationale des Syndicats d'Exploitants agricoles" (FNSEA) and a scientific Institute, the "Institut de Protection et de Sûreté Nucléaire" (IPSN).

2. Motivation and objectives

For many years, the FNSEA, on behalf of its adherents, has been giving attention to the potential consequences of the development of nuclear activities in the agricultural field : we must keep in mind that this development is not taking place in cities but in rural areas.

Before the Chernobyl accident, agriculture and nuclear activities coexisted without any major difficulty. But after the accident, the situation changed completely as the farmers were suddenly faced with a lot of confused, and sometimes contradictory, information to which they did not understand much except that they encountered difficulties for the sale of their productions.

FNSEA leaders were soon convinced that, in this field as well as in others, good information may avoid a regressive attitude and allows to face up possible problems. As a consequence, they decided to approach nuclear experts, and particularly a number agronomists or veterinaries working at IPSN.

They wrote together the above mentioned document which was greatly appreciated, thus demonstrating a serious need.

3. The audience

The first category of people concerned by the document were FNSEA adherents. Farmers are familiar with the environment, its strength and its weakness : in addition, they are realistic and well aware that sometimes one must compromise with events such as natural agressions they are often faced with. We must also remember that farmers were the main victims of the Chernobyl accident.

But the audience could be broader, including those directly or indirectly concerned with agriculture (administrative bodies, public or professional organizations, foodstuff industry...) or with environmental problems, or the medias (especially the scientific medias), the teaching profession... and any citizen anxious to be informed and fully aware of environment as his lifeplace and agriculture his lifesupport.

4. The objectives

The main objective of the document appears in its title : "how to react in the case of an accident". It intends to provide information allowing, in the case of an emergency, to take any initiative apt to mitigate the radiological consequences of the accident and later on cleverly to apply the protective measures recommended or laid down by the competent authority.

This information can help the whole agricultural community to preserve their interests and avoid being overtaken by a situation that is sometimes used by some people for reasons having nothing to do with radiation protection.

To reach these objectives, it is necessary both to assimilate the information and to understand it as fully as possible. As a consequence, the document has also an explanatory objective, especially concerning radionuclide transfer into the environment, the resulting contamination of agricultural products and the consequences of the transformations and uses of theses products.

It also appeared necessary to present a more general information on radioactivity and radiation effects, nuclear accidents (accidental risks associated with the use of nuclear energy, the main radionuclides involved), organization of intervention and protective measures).

5. The content

The structure of the document results from our objectives, i.e. :

- basic knowledge
- consideration about nuclear accidents (including intervention)
- contamination of the different components of the environment (air, water, soil...)
- contamination of the agricultural products (vegetables, fruits, cereals, milk, meat...)
- pratical exemples.

In order to make reading of this rather big document easier, it was decided :

- to avoid the use of unusual word and to give a definition of the specific technical terms,
- to use a system of references betwen complementary or similar parts of the document,
- to make each topic self-sufficient, so that the reader can read only he is interested in,
- to use a standardized presentation whenever possible, for instance :

  . how contamination occurs ?
  . what is the resulting risk ?
  . how to get protected ?

- to balance theoretical explanations and practical applications.

Nevertheless, this approach cannot avoid some traps, especially associating simple explanations and scientific accuracy. Besides It is sometimes difficult to present aspects which are still in evolution or not fully agreed upon.

6. Conclusion

It would be illusive to believe that such an ambitious objective can be reached, in a field still rather new, where all is not always clear and fully known and where emotional aspects sometines prevail over rationality.

•To be objective, to avoid any catastrophic approach but also any lenitive trend, to try to be clear and well understood : such was authors' concern. It will be very useful to know whether they have been sucessful.

WHOLE BODY MONITORING OF POST-CHERNOBYL BODY BURDENS IN RUSSIA

P.Hill and R.Hille
Forschungszentrum Jülich GmbH, ASS, Postfach 1913, D-517 Jülich

Summary

The body content of radionuclides of more than 160.000 persons in the RSFSR has been examined during summer 1991. An area of approximately 10.000 km² extending from close-to-Moscow Tula up to Zlynka at the Byelorussian border and such including regions of low and high contamination levels in the districts of Brjansk, Kaluga, Orel and Tula has been covered in the survey. Up to 7 mobile whole-body counting laboratories have been operational with a total number of 20 counters. More than 100 german specialists have participated in the task group. Each person examined received an official certificate giving the activity of the Cs-137/134 mixture in the body and a judgement on the radiation risk involved. Observed activity levels were generally much less and in no case higher than the annual limit of intake acceptable for professional radiation workers. Only for less than 1% of the people examined the results suggest the necessity of a further surveillance by health physicists and physicians similar to the regular medical examination professional radiation workers undergo.

1.Introduction

The aid programme [1] of the German 'Federal Ministry of Environmental Affaires, Protection of Nature and Reactor Safety' for regions of Russia affected by fall-out from the Chernobyl atomic power station included in 1991 investigations on both environmental [2] and body burdens [3,4]. The aim was an objective information of the population in order to reduce unnecessary fears and to remove psychological strain. People with high burdens have been identified and been reported to the Russian Health Service for future monitoring and medical care. In all of the more than 110 settlements and 12 counties covered by the survey the intakes of Cs-137/134 were generally much less and in no case higher than the annual limit of intake for professional radiation workers.

2.Equipment

Whole-body counters and incorporation monitors had to be chosen, which allowed mobile operation and had a reasonable limit of detection (LOD) for Cs-activities (appr. 1 kBq) even at the counting times of typically not more than 1 min necessary to maintain a sufficiently high throughput of people. Three trailers have been equipped with two Canberra FASTSCAN counters and two Herfurth H 13010 WB incorporation monitors each. An additional two of Nuclear Enterprises's QBM-1 incorporation monitors have been put into each of 4 Mercedes 609D vans.

2.1 Model 2250 FASTSCAN Whole Body Counter

The two large NaJ(Tl)- detectors (7.6 cm x 12.7 cm x 40 cm) of the Canberra FASTSCAN WBC allow the safe identification of incorporated γ- emitters. The efficiency is quite insensitive to the size and shape of the person monitored in a standing position. We have been able to measure reliably the body burdens as well of children of 12 kg weight as of adults using the same efficiency curve by slightly adjusting the position of children relative to the detectors by means of a turned over paper basket. The steel shielding with its attenuation factor of 10 for the Cs-137's 662 keV - line reduces background very efficiently. With one minute of measuring time a LOD of 260 Bq Cs-137 for adults and of 150 Bq Cs-137 for toddlers was obtained. Only in very high background areas the LOD increased by a factor of two.

2.2 Herfurth H 13010 WB Incorporation Monitor

The H 13010 WB uses a large 50 cm x 80 cm x 10 cm plastic scintillator in a bed configuration. A 5 cm lead shielding suppresses direct background radiation . The incorporation monitor has been calibrated for a realistic Cs-137/134 mixture. We used the monitors mainly for adults. Using real people with high body burdens to calibrate the H 13010 WB proved to be superior to a phantom calibration. A LOD of 500 Bq seems to be realistic in practical operation.

2.3 Nuclear Enterprises QBM-1

The QBM-1 uses two plastic scintillation detectors in a chair configuration. Originally designed for monitoring lung activities a shape- dependent correction factor had to be applied to account for both the whole-body distribution of Cs and the small admixture of Cs-134. The QBM-1 is more sensitive to environmental background than the FASTSCAN or the H 13010 WB are. The LOD of typically 1.1 kBq is higher than expected from counting statistics partly due to background fluctuations.

2.4 Quality control

Quality control included regular background checks as well as calibration checks with radioactive sources. Calibration of the incorporation monitors relatively to the FASTSCAN systems has also been checked by measuring the same people on several counters. An intercalibration with a russian phantom used in an IAEA- intercomparison [5] had been sucessfully performed (3).

3.Results

The open and objective communication of results obtained lead to a high acceptability of our programme in the local population. In some rural counties the body burdens of up to 70% of the population have determined and included in the Russian dose registry. A typical days work consisted of 500 (up

Table 1:Results on whole body monitoring of Russian People 1991

district	county	class 1	class 2	class 3	total
Brjansk	Gordejewka	4879	1752	270	6901
Brjansk	Klincy	43024	3029	924	46977
Brjansk	Krasnaja Gora	5727	1151	193	7071
Brjansk	Novosybkov	5422	461	45	5928
Brjansk	Zlynka	6677	1476	162	8315
Brjansk	Starodub	5124	20	0	5144
Kaluga	Chwastowitschi	9554	377	21	9952
Kaluga	Ludinowo	25189	42	9	25240
Kaluga	Shistra	9939	43	0	9982
Kaluga	Uljanovo	8136	259	4	8399
Orel	Bolchov	1446	1	0	1447
Tula	Arsenjewo	4953	0	0	4953
Tula	Plawsk	4060	0	0	4060
Tula	Uslowaja	18661	3	0	18664
		152791	8614	1628	163033
Brjansk		70853	7889	1594	80336
Kaluga		52818	721	34	53573
Orel		1446	1	0	1447
Tula		27674	3	0	27677
RSFSR		152791	8614	1628	163033

district	county	class 1	class 2	class 3
Brjansk	Gordejewka	70,7 %	25,4%	3,9 %
Brjansk	Klincy	91,6 %	6,4 %	2,0 %
Brjansk	Krasnaja Gora	81,0 %	16,3 %	2,7 %
Brjansk	Novosybkov	91,5 %	7,8 %	0,8 %
Brjansk	Zlynka	80,3 %	17,8 %	1,9 %
Brjansk	Starodub	99,6 %	0,4 %	0,0 %
Kaluga	Chwastowitschi	96,0 %	3,8 %	0,2 %
Kaluga	Ludinowo	99,8 %	0,2 %	0,0 %
Kaluga	Shistra	99,6 %	0,4 %	0,0 %
Kaluga	Uljanovo	96,9 %	3,1 %	0,0 %
Orel	Bolchov	99,9 %	0,1 %	0,0 %
Tula	Arsenjewo	100,0 %	0,0 %	0,0 %
Tula	Plawsk	100,0 %	0,0 %	0,0 %
Tula	Uslowaja	100,0 %	0,0 %	0,0 %
Brjansk		88,2 %	9,8 %	2,0 %
Kaluga		98,6 %	1,3 %	0,1 %
Orel		99,9 %	0,1 %	0,0 %
Tula		100,0 %	0,0 %	0,0 %
total		93,7 %	5,3 %	< 1,0 %

to 820) examinations on a trailer and 300 (up to 562) examinations on a van.

Every person examined obtained an official certificate testifying the body burden of Cs-137/134 and signed by a german specialist. The results have been grouped in three classes. Class 1 corresponds to body burdens of less than 7 kBq (4 kBq), class 2 of more than 7 kBq (4 kBq) but less than 25 kBq (15 kBq), class 3 of more than 25 kBq (15 kBq), numbers in brackets being valid for children. The class assignment was indicated on the certificate, too.

The body burdens have been shown to be small in more than 99% of the all cases (class 1 and class 2 in table 1). Health risks due to contamination of nutrition can be excluded for this part of the population. Only less than 1 % of the measurements yielded body burdens in class 3. Most of these cases have been found close to the Byelorussian border, where high environmental burdens exist. All body burdens observed have been assumed to be due to a continuous intake of radionuclides. Further assuming , that only a third of the annual dose commitment is from internal burden, it is concluded that not in a single case the annual dose commitment exceeded the limit of 50 mSv professional radiation workers are still permitted to receive. Future monitoring and surveillance by physicians and health physicists similar to the regular medical examinations professional radiation workers undergo are suggested for all those having a class 3 body burden.

Table 2: average annual internal dose 1991 of adults in some villages of Klincy county

settlement	number of adults examined	average annual internal dose [mSv/a]
Unetscha	203	1,7
Beresowka	99	1,3
Saretschje	106	1,1
Weprin	164	0,99
Uschtscherpje	733	0,72
Blisna	69	0,65
Tulukowschina	180	0,55
Guta-Karetzkaja	346	0,52
Roshnij	538	0,52
Lopatni	518	0,38
Teremoschka	112	0,29
Subbowitschi	35	0,27
Olchowka	675	0,26
Pestschanka	99	0,20

Using a dose factor of 0.040 mSv/kBq*a internal doses for adults being older than 20 years have been calculated. All settlements in the county of Klincy with an average annual internal dose of 0.2 mSv/a and more are listed in table 2. Results less than LOD have been included with the value of the LOD in the calculation of the mean. The survey being continued in 1992 some of those villages are included in a small medical programme [6]. Special care will be taken to monitor additionally individual external doses for a selected group of people.Radiation exposure of the Russian population after the Chernobyl accident is further discussed in [7].

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Part 5: Panel Discussion

Panel-Discussion

To conclude the meeting, a **panel discussion** was organised. The chairman of the panel was *Claude BIRRAUX* (Deputé de Haute Savoie/France) and the vice-chairman was *Bernard MICHAUD* (Head of the Radiation Protection Division of the Federal Office of Public Health/Switzerland). Three main topics were chosen, each introduced by a speaker from each of the two organising societies. As it is impossible to report literally all the contributions to the discussion, we have tried to summarise the most important statements on the topics.

The first of the topics, the **Environmental Impact of Nuclear Installations,** was introduced by *Serge PRETRE* (Congress President and Head of the Radiation Protection Division of the Nuclear Safety Inspectorate/Switzerland) and *André LE CORRE* (Département Sécurité Radioprotection Environnement of Electricité de France).

In western European countries, radioactivity discharges from nuclear installations to the environment, and the radiation exposure of the workers at these plants and of the surrounding population, have dropped considerably in the last ten years. Persons living near commercial west European nuclear power stations accumulate annual radiation doses of only some μSv. Radioactivity discharges from nuclear fuel reprocessing plants are still higher than those from commercial nuclear power stations, but have also diminished, due to substantial technical improvements and new regulations. Except for some regions - especially in eastern Europe, where the Chernobyl accident or nuclear weapon production facilities or tests and other military installations have caused radioactive contamination - artificial radioactivity in the environment shows very low values and is mostly not detectable. The environmental radioactivity surveillance has improved and modern technology creates highly sensitive automatic environmental monitoring networks. But the increased public concern about radioactivity and radiation should not lead - after the new ICRP-60-Recommendations - to a further lowering of dose and concentration limits. What we need, however, is to improve our knowledge of radioecological processes in the environment, comprising the whole nuclear cycle - from uranium mines to nuclear waste disposal and plant decommissioning. Another field where greater effort is necessary are the relations between radiation and health effects, i.e. further epidemiological studies should be undertaken, including an improved dosimetry, in particular that concerning radon and medical radiation exposure. Finally, western Europe should contribute to the technical and safety improvement of eastern European nuclear power stations and should therefore promote an enhanced technological and scientific transfer.

The second topic was entitled **Protection Measures for Nuclear Accidents and International Co-operation,** with *Hansheiri BRUNNER* (Secretary of the Fachverband, Swiss National Emergency Centre) and *Philippe VESSERON* (Directeur de l'Institut de Protection et de Sureté Nucléaire/France) as speakers.

In case of a nuclear accident, the information of the authorities and the population, the emergency measurement programmes, and the choice of the appropriate pro-

tective measures in a distance of up to some km from where the accident took place, are prepared in every country by co-operation of national and local authorities. The Chernobyl accident demonstrated, however, the necessity of certain emergency programmes up to distances of some 200 km. International co-operation became urgent. Therefore, bilateral early notification treaties have been signed between many of the western European countries, as well as suitable international treaties within the EC and the IAEA member states, to improve a rapid information exchange in case of a severe nuclear accident. But this is just the beginning and the contacts have to be further intensified to include local authorities, the health physics teams of national laboratories and the national emergency centres, particularly regarding intervention levels, emergency measurement programmes, automatic environmental monitoring networks, protective measures, and the assessment of the radiological impact on population and environment of such accidents. To increase this co-operation, bilateral scientific commissions have been established between France and Germany, France and Switzerland, and Germany and Switzerland. The aim is to harmonise the radiological criteria for nuclear accidents and the calculation models for protective measures, to open national data bases on environmental radioactivity (especially those from the automatic networks) to the health physicists and authorities of the neighbour countries, and finally to organise international exercises and prepare scientific and epidemiological programmes that would be carried out after such accidents.

Finally, the subject of **Public Relations in Radiation Protection** was discussed with *Rupprecht MAUSHART* (Publications Officer of the Fachverband/Germany) and *Pierre-Henri GERGONNE* (journalist/France) as speakers.

It came out that health physicists sometimes feel unhappy about mass media when informing the public in the field of radioactivity, the environmental impact of nuclear facilities and radiation risks. Scientists definitely have complete information on all these questions but often are not able to translate their knowledge in a form that is understandable for journalists and the public. Much scientific information, for example on radiation risk or cancer probability, is difficult to explain to someone without a minimum of mathematical knowledge. In many cases scientists are members of an authority, a research institute or a power generating company and therefore seem less credible for the representatives of the mass media than the so called "independent scientists". These institutions often have their own public relations departments impeding direct contact between their scientists and the mass media. On the other hand journalists have to "sell" their information and they sometimes pay more attention to spicy details than to the intention and the real content of a message. This kind of unintentional misunderstanding can be avoided by a constant dialogue between health physicists and journalists with the aim of an objective and understandable public information on risks and benefits of every application of radioactivity and radiation. Health physicists have to remember that their main duty is to show up risks and dangers of radiation, where there are some, comparing them to the risks of our daily life and preventing panic reaction where there is no danger.

H.V.

Part 6: Industrial Exhibitors

Amersham Buchler GmbH & Co

Gieselweg 1
D-3300 Braunschweig

Tel. ++49 5307 206 221
Fax. ++49 5307 206 272

Alnor Oy

PO Box 506
SF-20101 Turku

Tel. ++358 21 30 87 00
Fax. ++358 21 30 93 02

ARIES

44 bis, Bd Félix Faure
F-92320 Chatillon

Tél. ++33 1 46 57 41 71
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Baechli Instruments

Chrämerhoger 5
CH- 8910 Affoltern am Albis

Tel. ++41 1 761 71 28
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Hahnstrasse 70
D-6000 Frankfurt am Main 71

Tel. ++49 69 663 010
Fax. ++49 69 663 5921

FAG Kugelfischer Georg Schäfer KGaA

Postfach 1660
D-8520 Erlangen

Tel. ++49 9131 9 09 364
Fax. ++49 9131 90 92 05

Genitron Instruments GmbH

Heerstrasse 149
D-6000 Frankfurt am Main 90

Tel. ++49 69 768 11 44
Fax. ++49 69 765 327

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CH-8500 Frauenfeld

Tel. ++41 54 720 30 30
Fax. ++41 54 720 30 50

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D-2000 Hamburg 50

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Fax. ++49 40 89 69 42 56

Hörmann Entwicklungs GmbH

Hauptstrasse 45 - 47
D-8011 Kirchseeon b. München

Tel. ++49 8091 52-0
Fax. ++449 8091 1275

Laboratorium Prof. Dr. Berthold GmbH & Co

Calmbacher- Strasse 22
D-7547 Bad Wildbad 1

Tel. ++49 7081 177-204
Fax. ++49 7081 177-166

Mini-Instruments Ltd.

8 Station Industrial Estate
Burnham on Crouch, Essex
England CMO 8RN

Tel. ++44 621 78 32 82
Fax. ++44 621 78 31 32

Münchner Apparatebau
für elektronische Geräte GmbH

Mehlbeerenstrasse 2
D-8028 Taufkirchen 1

Tel. ++49 89 61 2003 15 /16/17
Fax. ++49 89 61 2003 35

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Chemin des Clos
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F-38244 Meylan CEDEX

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CH-1004 Lausanne

Tel. ++41 21 25 24 23
Fax. ++41 21 25 23 53

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Target systemelectronic gmbh

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D-5650 Solingen 1

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Fax. ++49 212 20 10 45

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CH-5040 Schöftland

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Fax. ++41 64 81 33 88

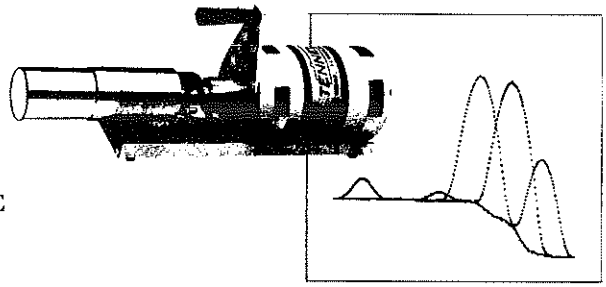
Veenstra Instruments BV

Madame Curiewei 1
NL-8500 AC Joure

Tel. ++31 5138 16 964
Fax. ++31 5138 16 919

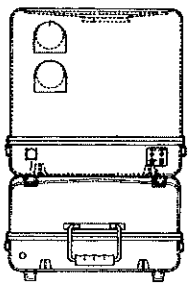
SPECTROMETRIE GAMMA PORTABLE
PORTABLE GAMMA SPECTROSCOPY

- TISA carte PC pour sonde NaI
- *TISA PC board for NaI crystal*
- ISA carte PC pour détecteur Ge
- *ISA PC board for Ge detector*



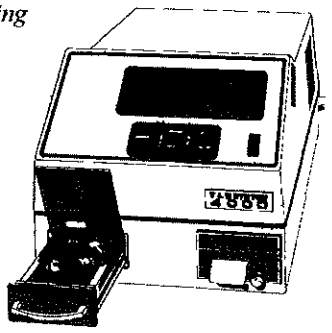
SYSTEME DE MESURE EMBARQUE
MOBILE MEASUREMENT SYSTEM

Deux acquisitions simultanées :
Two simultaneous acquisitions :
- Débit de dose Gamma ambiant
- *Ambiant Gamma dose rate*
- Spectrométrie gamma automatique
- *Automatic Gamma spectroscopy*



BALISE AUTOMATIQUE DE PRELEVEMENTS
AUTOMATIC SELF-CONTAINED SAMPLING DEVICE

- Mesure et enregistrement permanent du débit de dose
- *Permanent measurement and recording of Gamma dose rate*
- Prélèvements d'air et d'eau de pluie
- *Air and precipitations sampling*
- Déclenchement sur consigne horaire ou seuil de débit de dose
- *Sampling start according to time or dose rate level presetting*

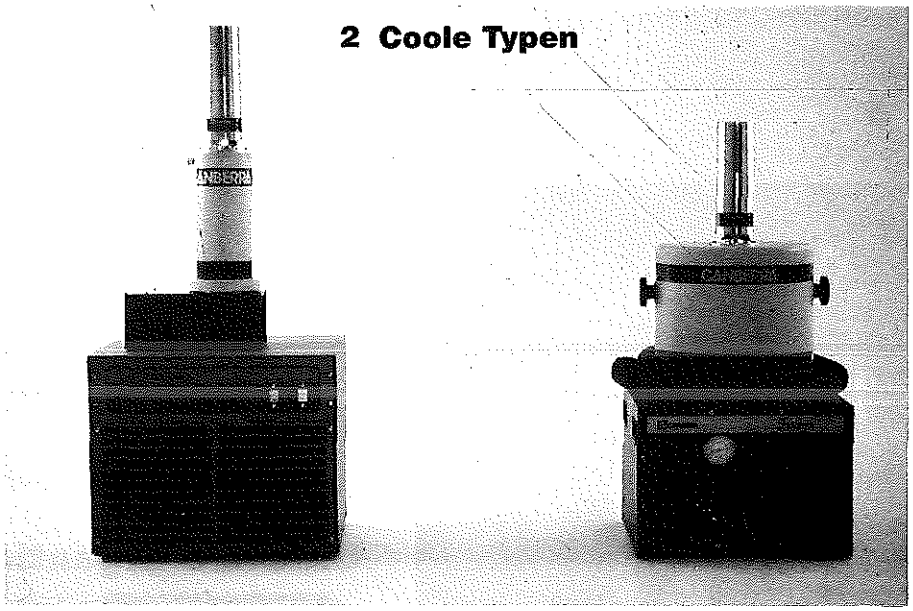


DOSIMETRIE TLD
TLD DOSIMETRY

- Dosimètres et lecteurs HARSHAW/QS
- *HARSHAW/QS dosimeters and readers*
- Lecteur passeur d'échantillons DOSIMATIC
- *Reader with "DOSIMATIC" sample changer*
- Machines de conditionnement de cartes et de badges
- *Computer controlled machines for handling of cards and holders*

44 bis, Boulevard F.Faure 92320 CHATILLON (France)
Tél. (33) 1 46 57 41 71 Télécopie. (33) 1 46 56 69 39

2 Coole Typen



Cryolectric

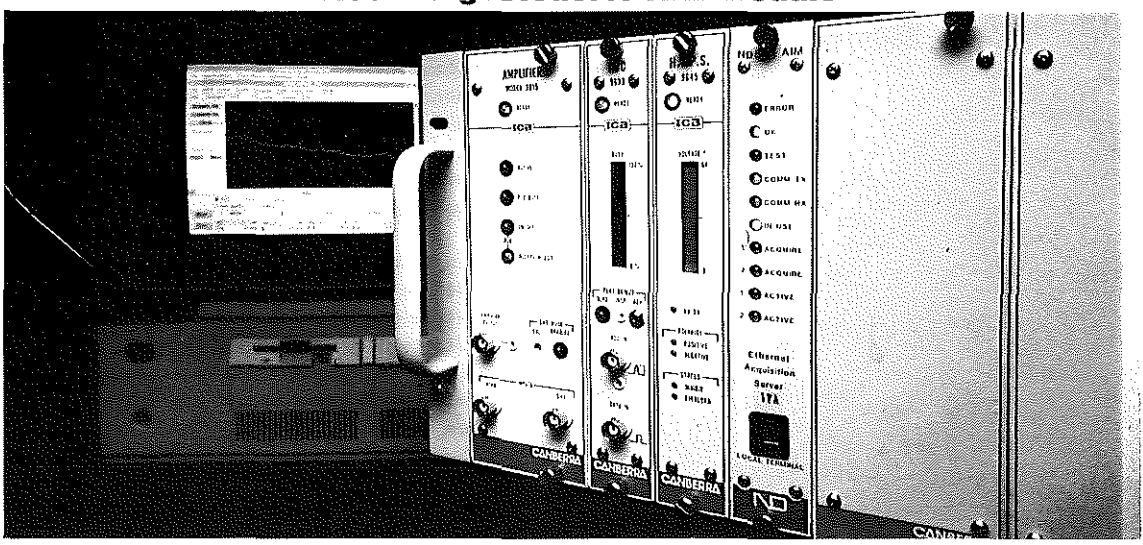
- Helium Kühlaggregat mit flexibler Schlauchverbindung zum Detektorkopf mit max. 30 m Länge
- Mehrere Detektoren können von einem Kompressor versorgt werden
- Der Detektorkopf arbeitet auch bei widrigsten Umgebungsbedingungen
- Robuste Industrieausführung
- Geringste Wartungsintervalle

Freolectric

- Zuverlässige und kostengünstige Detektorkühlung für den Laborbereich
- Einfache Adaption an vorhandene Bleiabschirmungen
- Leises und erprobtes Industriedesign

EINE ALTERNATIVE ZUR STICKSTOFF-KÜHLUNG.

Rechnergesteuerte NIM-Module



ICB-NIM – die neueste Entwicklung von Canberra

Computer kontrollierte NIM Module für die ferngesteuerte Datenerfassung. Zur Qualitätssicherung werden alle Einstellparameter mit den Meßdaten gespeichert.

- CI-9615 Spektroskopie Verstärker
- CI-9645 Detektor Hochspannung
- CI-9633 Analog Digital Wandler
- AIM Ethernet Vielkanalanalysator-Modul

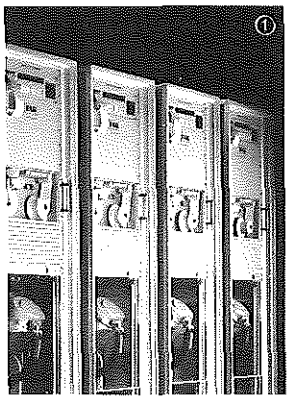
CANBERRA PACKARD

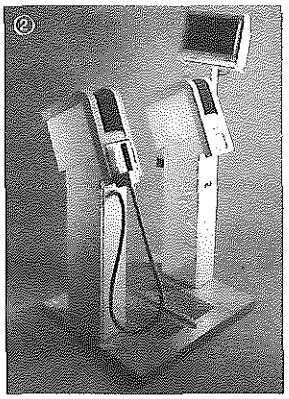
Canberra-Packard GmbH
Hahnstraße 70
6000 Frankfurt / Main 71

Telefon: (069) 6 63 01-0
Telex: 4 16 037
Telefax: (069) 6 66 59 21

Strahlung sicher bewerten

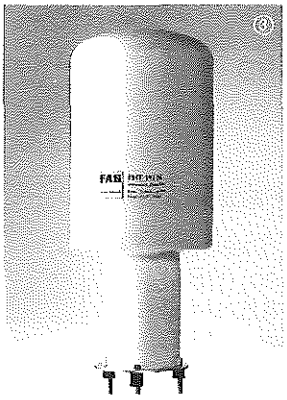
Der FAG Erzeugnisbereich Strahlen-Meßtechnik entwickelt und produziert Meßgeräte zum Nachweis künstlicher ionisierender Strahlung. Durch den Einsatz dieser Geräte und die dadurch ausgelöste frühzeitige Warnung lassen sich u.a. Sicherheitsstandards verbessern und Gesundheits- und Umweltrisiken mindern.
Programm: Tragbare digitale Dosis-, Dosisleistungs- und Kontaminationsnachweisgeräte. Überwachungs- und Kontrolleinrichtungen, Labormeßgeräte, automatisch arbeitende Luftüberwachungsanlagen (z.B. in den Stationen des Deutschen Wetterdienstes, des BMU, des LFU Bayern usw.), eigenständige Elektronik- und Software-Entwicklungen sowie Staubmeßgeräte für Immissionen, Emissionen und Abgaspartikel.

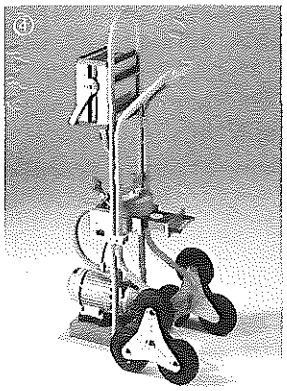




① **Jodmonitor FHT 1700,** kombiniert mit Aerosol-Schrittfilteranlage FHT 59 S (Ausführung für das Meßnetz des Deutschen Umweltbundesamtes). Registriert werden die aktuelle Aktivitätskonzentration für I-131 und aerosolgebundene Spaltprodukte sowie die akkumulierte I-131-Aktivität.

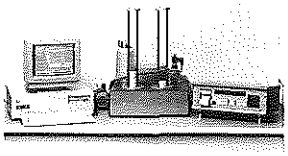
② **Hand-Fuß-Kleider-Monitor FHT 65 P** zur Personenüberwachung am Kontrollbereich. Getrennte oder simultane Messung von Alpha- und Beta-Kontaminationen. Mit einem handelsüblichen Personenidentifikations-System kann der Monitor direkt in den Personenfluß eingebunden werden.





③ **Hochleistungs-Ionisationskammer FHT 191 N** mit integrierter Intelligenz. Durch spezielle Schaltkreise ist ein weitgespannter Meßbereich von 10 nSv/h bis 10 Sv/h zu verzeichnen. Die Dosisleistung wird auch bei gepulsten Strahlungsquellen korrekt ermittelt.

⑤



⑤ **Wassermonitor FHT 1900** für quasi-kontinuierliche Probenentnahme. Gegenüber bisher üblichen Alpha- und Beta-Überwachungssystemen ist mit dieser Meßanordnung eine Verbesserung der Nachweisempfindlichkeit für Alpha- und Beta-Strahler um den Faktor 100 bis 1000, je nach Meßzeit und Nuklid, zu verzeichnen.

④ **Mobiler Monitor FHT 1659 L** für die Überwachung der luftgetragenen aerosolgebundenen Radioaktivität an wechselnden Einsatzorten. In Verbindung mit einem Personal Computer ist eine zentrale On-line-Datenerfassung mit Darstellung des Meßwertverlaufs möglich.

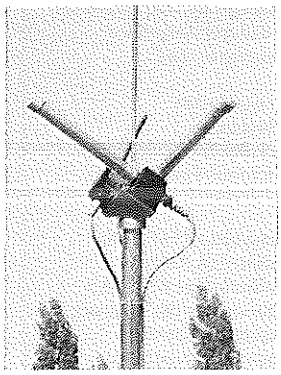
FAG

ahlen-Meßtechnik

FAG Kugelfischer Georg Schäfer KGaA

Erzeugnisbereich Strahlen-Meßtechnik
Postfach 1660 · W-8520 Erlangen · Germany · Telefon (0 91 31) 90 90· Telefax (0 91 31) 909 205

1982 - 1992: TEN YEARS OF EXCELLENCE IN SELECTED FIELDS OF RADIATION MEASUREMENT

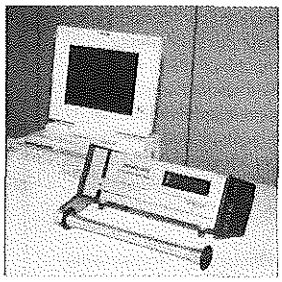


Creativity

- ❑ New ideas honoured with awards and patents
- ❑ Participation in public R&D programmes
- ❑ Flexible engineering service
- ❑ Unorthodox approaches

Top Technology

ASIC-design ❑
Thick-film hybrids ❑
CAE-tools ❑

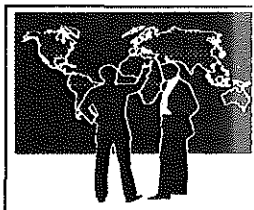


Innovative Product Lines

- ❑ MINI-MONITOR, MIRA-661
- ❑ GeoSCAN, GammaTRACER
- ❑ IRMA, Janus Head, ELIS-Sensor
- ❑ ATMOS-12PX, AlphaGUARD
- ❑ E-PERM
- ❑ RadNUK

International Partnership

Gammadata (Sweden) ❑
Rad Elec Inc. (USA) ❑
Positron (Ukrainia) ❑
J.S.Bland Assoc. (USA) ❑

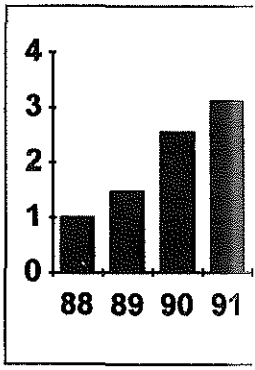


Scientific Ambition

- ❑ Cooperation with Universities
- ❑ Scientific publications
- ❑ Field studies
- ❑ Workshops
- ❑ Seminars

Dynamic Growth

. . . result of all activities



Genitron Instruments GmbH, Heerstraße 149, D-6 000 Frankfurt 90,
Tel.: 0049-(0)69 768-1144, Fax: 0049-(0)69 765-327

GOTELE

VICTOREEN

Messgeräte für den Strahlenschutz

Dosis
Dosisleistung
Kontamination
Umgebungs-
überwachung
Personenschutz
Rn-Messungen

ildung:
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l RP-1 Pancake Probe

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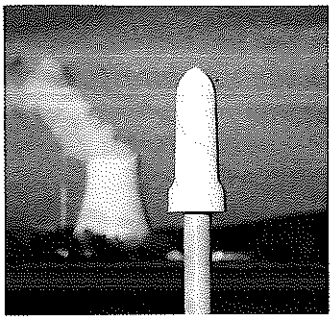
Nuclear Radiation and Environmental Measuring System

From a Sensor to an Integrated Networ



Detector Probe

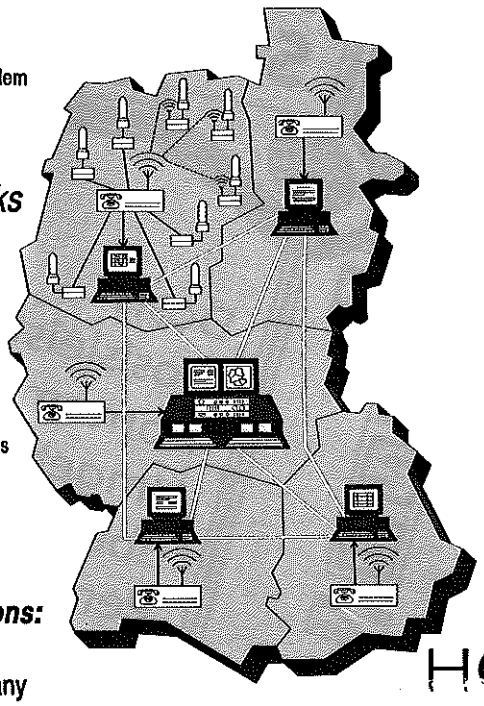
Probes for dose rate detection with different sensitivity

Intelligent probes with variable number of detectors
Interface alternatively RS 485, RS 232C, FSK modem or radio data transmission

Monitoring Networks

Conception and realisation of integrated monitoring networks for:
- monitoring of nuclear facilities
- wide area monitoring systems

Modular communication units



Monitoring Stations

Data gathering systems for connection of gamma probes with additional interfaces for further sensors, e.g. meteoro air and waste water analysis

Software

Data communication and presentation software for MS-DOS and UNIX systems

Mobile Systems

General purpose vehicles with individual measuring and monitoring devices

HÖRMANN installations:

WADIS, Germany
about 2100 monitoring units
Integrated Network Spain
900 Intelligent gamma probes
MADUK, Switzerland
58 monitoring units
Integrated Network Portugal
15 monitoring units
NPP Pilgrim Station, Plymouth, Mass. USA
15 monitoring units

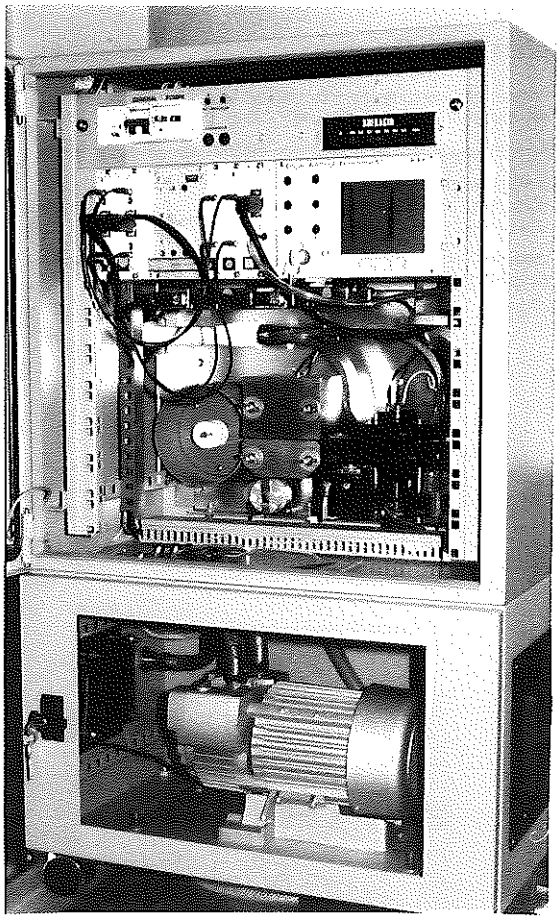
HÖRMANN
Entwicklungs GmbH

Hauptstraße 45-47
D-8011 Kirchseeon
Germany
Tel. 08091/52-261
Fax 08091/1275

c:\designer\anz_1

REAL TIME α and β

MONITORING IN AIR



BAE

ENVIRONMENT α-ß AEROSOLS MONITOR

- **Direct and delayed measuring heads**
- **Real time gamma compensation**
- **Real time spectrometric Radon-Thoron compensations**
- **Self check by statistical validity calculations**
- **Spectrae display**
- **Network inter-connexion**

OTHER INSTRUMENTS :

BAB-A	:	Portable alpha-beta in air monitor
CST 28	:	Large area roads contamination monitor
DG5	:	Gamma source finder, very high sensitivity
MSA-P	:	Drive-through vehicle monitor

WASTE MONITORING MACHINES

---

zirst - bp 147 - 38244 meylan cedex - france
tél. 76.90.70.45
fax 76.41.04.36

NOVELEC S.A.

VACUTEC MESSTECHNIK

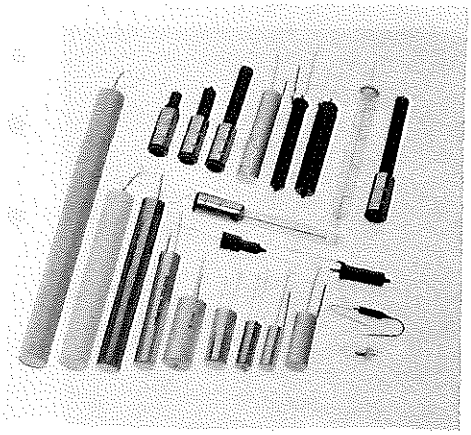
Vacutec GmbH
Dornblüthstraße 14
O-8021 Dresden
Tel.: 03 51 / 3 47 52 05
Fax: 03 51 / 3 50 85

GM-Counters

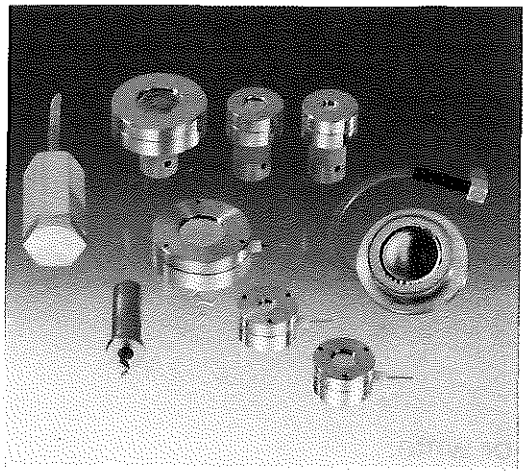
Proportional Counters

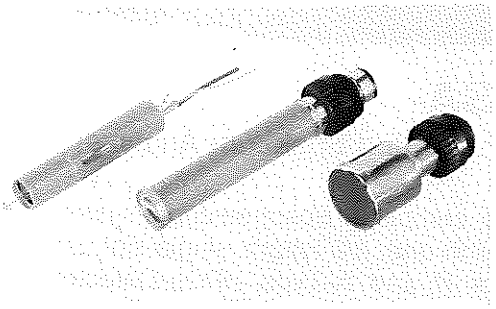
Ionisation Chambers

Si(Li) Semicontuctors



Vacutec* produziert seit über 35 Jahren Zählrohre, Ionisationskammern
und HLD's für Lehre und Forschung, Industrie und Medizintechnik.
Inbegriffen sind energiekompensierte Zählrohre für Umweltmessungen
in überregionalen Meßnetzen und Einzelgeräten.
Erfahrungen aus der Militärtechnik und Einsatz in einer Vielzahl von
PTB zugelassenen Erzeugnissen zeugen von hoher Qualität und Zuverlässigkeit.
Wir lösen mit unseren Detektoren gern Ihre Meßaufgaben.





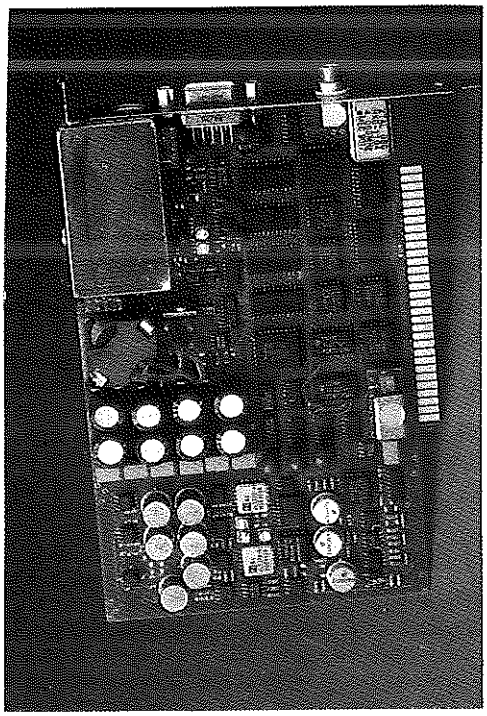
*ehemals ein Teil von
Robotron Meßelektronik Dresden

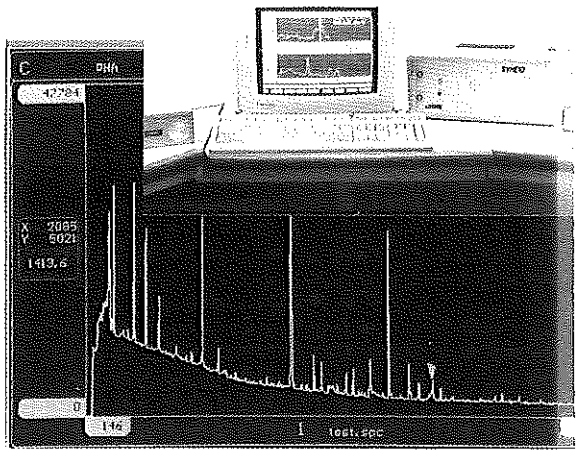
VACUTEC MESSTECHNIK

Vacutec GmbH
Dornblüthstraße 14 · O-8021 Dresden
Tel. 051 / 3475205 · Fax 051 / 35085

target

- Multichannel Analyzer Syste
- NIM Electronic
- Detectors
- Data Systems
- Special Turnkey Application
  - IN-SITU Gamma Spetroscopy
    - Waterproof, Ruggedized, Battery Powered
    - Integrated Quantitative Analysis
  - Foodmonitoring
  - Contamination Control System





ISA-

Integrated **S**pektrum **A**nalyzer

A complete Spectros-copy system, all in one, 1/2 length, single slot card for P and N-Type HPGe-Detectors as well as Si(Li), CdTe, NaI(TL) and other detectors.

For additional information call or write:

target

systemelectronic gmbh

Kölner Str. 99
W-5650 Solingen
Tel.:0212/201041
Fax:0212/201045
Germany

NUCLEAR ELECTRONICS. DATA SYSTEMS, SOFTWARE

List of Participants

SWITZERLAND

CHLI Anton	BÄCHLISTRUMENTS Chrämerhoger 5 CH-8910 AFFOLTERN am Albis
GGENSTOS Martin phys.	Hauptabteilung für die Sicherheit der Kernanlagen, Abt. Strahlenschutz CH-5232 VILLIGEN-HSK
RET Pierre Ing. ETS	Section Surveillance de la Radioactivité Office Fédéral de la Santé Publique Chemin du Musée 3 CH-1700 FRIBOURG
SER Beat Dipl. chem.	Kernkraftwerk Mühleberg Bernische Kraftwerke AG CH-3203 MÜHLEBERG
CHUNG Markus phys.	Paul Scherrer Institut Abt. Strahlenhygiene CH-5232 VILLIGEN-PSI
ET Claude Dr.	CERN SL Division Case Postale CH-1211 GENEVE 23
INNER Hansheiri Phys.	Nationale Alarmzentrale/GS-EDI Ackermannstrasse 26 Postfach CH-8044 ZÜRICH
GER Mario Dr.	Gruppe für Rüstungsdienste AC-Laboratorium Spiez AC-Zentrum CH-3700 SPIEZ
GERT Gerhard	Kernkraftwerk Gösgen-Däniken AG Postfach CH-4658 DÄNIKEN
TIER Franz Dr.	Hauptabteilung für die Sicherheit der Kernanlagen, Abt. Strahlenschutz CH-5232 VILLIGEN-HSK
N David Dr.	Dammstrasse 5 CH-5400 BADEN
MENT René s-Directeur	Entreprises Electriques Fribourgeoises 25, Boulevard de Pérolles CH-1701 FRIBOURG
RNECKI Jerzy Dr.	Pestalozzistrasse 1 CH-5200 WINDISCH
TWYLER Jean-Jacques Journaliste	Wabernstrasse 34 CH-3007 BERN
RERI Giovanni	Sektion Überwachung der Radioaktivität Bundesamt für Gesundheitswesen Chemin du Musée 3 CH-1700 FRIBOURG
CHER Klaus Dr.	COLENCO Power Consulting AG. Mellinger Str. 207 CH-5405 BADEN
Y Carl-Heinz	Roman Spiess Strahlenschutz & Notfallplanung CH-5316 GIPPINGEN
LER Erwin Dr. sc. nat.	Kantonales Labor Zürich Fehrenstrasse 15 CH-8030 ZÜRICH

SWITZERLAND

GMÜR Kurt Dr.	Paul Scherrer Institut Abt. Strahlenhygiene CH-5232 VILLIGEN-PSI
GOBET Myriam	Section Surveillance de la Radioactivité Office Fédéral Santé Publique Chemin du Musée 3 CH-1700 FRIBOU
GOEBEL Klaus Dr.	28, Crets de Champel CH-1206 GENEVE
GOETZ Ernst	GOTELE AG Elektronik für Industrie & Wissenschaft Schönenhof CH-8500 FRAUENFELD
GURTNER André	Section Surveillance de la Radioactivité Office Fédéral Santé Publique Chemin du Musée 3 CH-1700 FRIBOU
HAGENLOCHER Ingeborg Dr.	NAGRA Hardstrasse 73 CH-5430 BADEN
HALLER Max Chemotechniker	Bernische Kraftwerke AG Kernkraftwerk Mühleberg CH-3203 MÜHLEBERG
HAUSWIRTH Gaston Ing. HTL	Paul Scherrer Institut Abt. Strahlenhygiene CH-5232 VILLIGEN-PSI
HERRNBERGER Volker Dipl. phys.	Paul Scherrer Institut Abt. Strahlenhygiene CH-5232 VILLIGEN-PSI
HÖFERT Manfred Dr.	CERN TIS-RP CH-1211 GENEVE 23
HUBER Otto Prof. Dr.	Beustweg 8 CH-8032 ZÜRICH
HUNZINGER Werner Dr.phil.	Gartenstrasse 7 CH-3074 MURI
JANETT Albin Dr.	Paul Scherrer Institut Abt. Strahlenhygiene CH-5232 VILLIGEN-PSI
JEANNET Pierre Dipl.-Phys.	Section Météorologie de l'environnement Station Aérologique Les Invuardes CH-1530 PAYERNE
JOHNER Christof	Physikinstitut der Universität Fribourg Chemin du Musé CH-1700 FRIBOURG
JOHNER Hans-Ueli Dr.	Physikinstitut der Universität Fribourg Chemin du Musé CH-1700 FRIBOURG
JUNGCK Matthias Dr.	Bundesamt für Gesundheitswesen Abteilung Strahlenschutz Postfach 2644 CH-3001 BERN
KHORIOUCHINE Dmitri Ing.	Universite d'Etat de VORONEJ Foyer St. Justin Route du Jura 3 CH-1700 FRIBOURG
LABNO Leszek	ABB Power Generation Ltd. Postfach CH-5401 BADEI
LERCH Pierre Prof. Dr.	Institut de Radiophysique appliquée Centre Universitaire CH-1510 LAUSANNE

SWITZERLAND

LEUPIN Andreas Dr.	Hauptabteilung für die Sicherheit der Kernanlagen, Abt. Strahlenschutz CH-5232 VILLIGEN-HSK
LÖRTSCHER Yves Ing. HTL	Paul Scherrer Institut Abt. Strahlenhygiene CH-5232 VILLIGEN-PSI
LOOSLI Hugo Prof. Dr.	Universität Bern Physikalisches Institut Sidlerstrasse 5 CH-3012 BERN
MAILLARD Stéphane	ABB Power Generation Ltd. Postfach CH-5401 BADEN
MARTI Jürg Dipl. Biol.	Bundesamt für Gesundheitswesen Abt. Strahlenschutz Postfach 2644 CH-3001 BERN
MICHAUD Bernard Dr.	Bundesamt für Gesundheitswesen Abt. Strahlenschutz Postfach 2644 CH-3001 BERN
MORF Jürg	Kernkraftwerk Gösgen-Däniken AG Postfach CH-4658 DÄNIKEN
MÜLLER Rudolf Dr.	Felixhäglistr. 4 CH-4103 BOTTMINGEN
MURITH Christophe Dr. dipl.phys.	Section Surveillance de la Radioactivité Office Fédéral de la Santé Publique Chemin du Musée 3 CH-1700 FRIBOURG
NÜNLIST Heinz	Kernkraftwerk Gösgen-Däniken AG Postfach CH-4658 DÄNIKEN
NYFFELER Francois Dr.	Université de Neuchâtel Inst. de Géologie, Groupe PROSPER 11 rue Emile Argand CH-2007 NEUCHATEL
OZIMEK Hans-Albert	GOTELE AG Schönenhof CH-8500 FRAUENFELD
PEROTTO Carlo Dipl.Ing.ETS	Bernische Kraftwerke AG Kernkraftwerk Mühleberg CH-3203 MÜHLEBERG
PILLER Georges Dr. dipl. phys.	Sektion Überwachung der Radioaktivität Bundesamt für Gesundheitswesen Chemin du Musée 3 CH-1700 FRIBOURG
PRETRE Serge Dipl.phys.	Hauptabteilung für die Sicherheit der Kernanlagen, Abt. Strahlenschutz CH-5232 VILLIGEN-HSK
RAUBER Dominique Dr.	Nationale Alarmzentrale GS-EDI Ackermannstrasse 26 Postfach CH-8044 ZÜRICH
RIBORDY Louis Dipl. Ing. ETHZ	Sektion Überwachung der Radioaktivität Bundesamt für Gesundheitswesen Chemin du Musée 3 CH-1700 FRIBOURG

SWITZERLAND

RICHON Bernard	COLENCO Power Consulting AG Mellingerstrasse 207 CH-5405 DAETTWIL
ROTH Martin	Berthold AG Adlikerstrasse 236 CH-8105 REGENSDO
SCHOBER Johannes Dipl.chem.	ABB Management AG Bruggerstrasse CH-5507 BADEN
SCHWARZ Georg Dr.	Eidg. Technische Hochschule Zürich Institut für Geophysik ETH-Hönggerberg CH-8093 ZÜRICH
STEFFES Francoise Dipl.Phys.	Oberdorf 435 CH-1735 GIFFERS
SULMONI-THOMI Dina Dipl. phys.	Nationale Alarmzentrale GS-EDI Ackermannstrasse 26 Postfach CH-8044 ZÜRICH
TUOR Stephan Dipl.Phys.ETH	ETH- IMS Auf der Mauer 2 CH-8001 ZÜRICH
TUYN Jan Willem Dr.	CERN TIS-RP CH-1211 GENEVE 23
ÜHLINGER Albert	ÜHLINGER-PFIFFNER AG NUCLEAR- UND MEDIZINTECHNIK Picardie-Strasse 3 CH-5040 SCHOEFTLAND
VALLEY Jean-Francois Dr.	Institut de Radiophysique appliquée Centre Universitair CH-1015 LAUSANNE
VAN DORP FRITS Dr.Ing.agr.	Nationale Genossenschaft für die Lagerung radioaktiver Abfälle Hardstrasse 73 CH-5430 WETTING
VÖLKLE Hansruedi Dr. dipl.phys.	Sektion Überwachung der Radioaktivität Bundesamt für Gesundheitswesen Chemin du Musée 3 CH-1700 FRIBOURG
VONARBURG Barbara Journalistin	Redaktion TAGES-ANZEIGER Postfach CH-8021 ZÜRICH
WEIDMANN Urs Dr.	Kernkraftwerk Beznau KBU-B CH-5312 DÖTTINGEN
WEITZE Hartmut Ing.	Kernkraftwerk Gösgen-Däniken AG Postfach CH-4658 DÄNIKEN
WERNLI Christian Dipl. phys.	Paul Scherrer Institut Abt. Strahlenhygiene CH-5232 VILLIGEN-PSI
WÜNSCHE-ETTER Silvia	Strahlenschutzsachv. CIBA-GEIGY AG ZSS K-127.690 Postfach 60 CH-4002 BASEL
ZEERLEDER, de Marc Ing. HTL	Bernische Kraftwerke AG Kernkraftwerk Mühleberg CH-3203 MÜHLEBERG
ZELLER Werner Dr.	Bundesamt für Gesundheitswesen Abteilung Strahlenschutz Postfach 2644 CH-3001 BERN

GERMANY

ADER Dieter Dipl. Ing.	E. Schmutz GmbH Römerstrasse 16 D-7858 WEIL a. RHEIN
AUER Bernhard W.	Bundesamt für Strahlenschutz Institut für Strahlenhygiene Ingolstädter Landstr. 1 D-8042 NEUHERBERG / München
AYER Bernhard Ing.	DEKONTA Industriedienste GmbH Tulpenstrasse 19 D-8031 MAISACH
AYER Anton Prof. Dr.	Bundesamt für Strahlenschutz Institut für Strahlenhygiene Ingolst. Landstr.1 Postfach 1108 D-8042 NEUHERBERG / München
EINLICH Uwe	Amersham Buchler GmbH & Co Gieselweg 1 D-3300 BRAUNSCHWEIG
ERG Heinz-Peter Dr.	Bundesamt für Strahlenschutz Albert-Schweitzer-Strasse 18 Postfach 10 01 49 D-3320 SALZGITTER 1
RNSTOCK Ronald Dr.	Beschleunigerlaboratorium der Uni und TU MÜNCHEN Hochschulgelände D-8046 GARCHING
SCHOF Werner Dr. jur.	Universität Göttingen Institut für Völkerrecht Platz der Göttinger Sieben 5 D-3400 GÖTTINGEN
TTNER Heinz	Technische Universität MÜNCHEN Sicherheitswesen/ Strahlenschutz Römerhofweg 67 D-8046 GARCHING
ORCHARDT Dieter pl. Phys.	Hahn-Meitner-Institut Berlin GmbH Abt. Strahlenschutz Glienicker Strasse 100 D-1000 BERLIN 39
RANDT Klaus-Dieter pl.-Ing.	Kraftwerks- und Anlagebau Berlin Holteistrasse 28 D-O-1035 BERLIN
RÜCHER Lothar Dr.rer.nat.	Siemens AG, Abt. KWU-R262 Berlinerstrasse 295-303 Postfach 10 10 63 D-6050 OFFENBACH / MAIN
: WITT Helmut Dr.rer.nat.	BRENK SYSTEM PLANUNG Heinrich-Allee 38 D-5100 AACHEN
ELLER Wunibald Dipl. Ing.	RWE - Energie AG Kruppstrasse 5 D-4300 ESSEN
OMKE Frank	Herfurth GmbH Beerenweg 6-8 Postfach 50 06 48 D-2000 HAMBURG 50
ORNSTETTER Paul	DORNIER GmbH Abt. VF1 402 Postfach 1420 D-7990 FRIEDRICHSHAFEN
'CK Werner Dipl. Ing.	Deutscher Wetterdienst Kaiserleitstr. 42 D-6050 OFFENBACH AM MAIN
'MANSKI Ulrich Dr.	Postfach 120213 D-6200 WIESBADEN

GERMANY

EBERBACH Friedrich Dr.	Ministerium f. Umwelt Rheinland-Pfalz Kaiser-Friedrich-Strasse 7 D-6500 MAINZ
ECKERL Helmut Dipl. phys.	SIEMENS AG Bereich KWU USt Hammerbacher Str. 12 + 14 D-8520 ERLANGEN
EIGENWILLIG G.-G. Dr.	SIEMENS AG KWU-N21 Berlinerstrasse 295-303 Postfach 10 10 63 D-6050 OFFENBACH am Main
EMRICH Hansjörg Dr.	Hessisches Ministerium für Umwelt Mainzerstrasse 80 D-6200 WIESBADEN
FEINHALS Jörg Dr.rer.nat.	Techn. Überwachungsvein Norddeutschland Gr. Bahnstrasse 31 D-2000 HAMBURG 54
FIEDLER Winfried	Wellhöfer Dosimetrie D-8501 SCHWARZENBRUCK
FINKE Elvira Dr.	Bundesgesundheitsamt Klinisch-Diagnostischer Bereich Waldowalle 117 D-O-1157 BERLIN
FISCHER Carsten	ADMINTEC GmbH Bayernstrasse 53 D-7530 PFORZH
FOLKERTS Karl-Heinz Prof. Dr.	Hochschule f. Technik u. Wirtschaft des Saarlandes Göbenstrasse 40 D-6600 SAARBRÜCKEN
FREIMANN I.	Canberra-Packard GmbH Hahnstrasse 70 D-6000 FRANKFURT AM MAIN
FRENZEL Eberhard Dr.	EG&G Labor Prof. Dr. Berthold Calmbachstrasse 22 Postfach 10 01 63 D-7547 WILDBAD 1
FRIEDLAND Eva-Maria	Hahn-Meitner-Institut Berlin GmbH Abt. Strahlenschutz Glienicker Strasse 100 D-1000 BERLIN 39
FRONZ Wolfgang Dipl.-Ing.	Landesanstalt für Immissionsschutz Nordrhein-Westphalen Wallneyerstrasse 6 D-4300 ESSEN 1
FUNSCH Benno Dipl.-Phys.	Landesanstalt für Umweltschutz Baden-Württemberg Hertzstr. 173 D-7500 KARLSRUHE 21
GAMMEL Reinhold Dipl- Ing.	Badenwerk AG Personalbetreuung Badenwerkstr. 2 Postfach 1680 D-7500 KARLSRUHE 1
GENRICH Volker Dipl. chem.	GENITRON Instruments GmbH Heerstrasse 149 D-6000 FRANKFURT 90
GÖTTLE Herbert Dipl.-Ing.	Ing-Büro Dipl.-Ing.Robert V. Krissel Schubertstrasse 4e D-7514 EGGENSTEIN
GRIESBACH Manfred Dr.-Ing.	Hessisches Ministerium des Innern und f. Europaangelegenheiten Friedrich-Ebert-Allee 12 D-6200 WIESBADEN

GERMANY

RIESE Klaus Dipl. phys.	KAB AG Berlin Allee d. Kosmonauten 32 D-O-1140 BERLIN
JGLHÖR Peter Dipl. Phys.	TÜV Bayern-Sachsen e.V. G2 ETS Westendstrasse 199 D-8000 MÜNCHEN 21
AMBECK Ludwig .sc.techn.	Wismut GmbH Techn. Ressort Jagdschänkenstrasse 29 D-O-9030 CHEMNITZ
ARRISCH Peter	Münchner Apparatebau GmbH Mehlbeerenstrasse 2 D-8028 TAUFKIRCHEN 1
AUG Thomas Dr.	Universität Tübingen Isotopenlabor Auf der Morgenstelle 24 D-7400 TÜBINGEN
AUSER Hans-Joachim ol.-Ing.	Regierungspräsident Düsseldorf Cecilienallee 2 D-4000 DÜSSELDORF
AUSKE Hans Dr.	Kerntechnische Hilfsdienst GmbH Am Schröcker Tor 1 D-7514 EGGENSTEIN-LEOPOLDSHAFEN
EDEMANN Wilfried	DBE Woltorfer Strasse 74 D-3150 PEINE
EINEMANN Günter . med.	Betriebsarzt PreussenElektra Tanklager Stadersand D-2160 STADE
EINEMANN Karl Dr.	Forschungszentrum Jülich Abt. Sicherheit & Strahlenschutz Postfach 1913 D-5170 JÜLICH-1
EINLE Walter	Kernkraftwerk Philippsburg GmbH Postfach 1140 D-7522 PHILIPPSBURG
EINTZ Werner Dr.	Reinhold-Schneider-Strasse 73B D-7500 KARLSRUHE
ENKE	Target Systemelektronik GmbH Kölner Strasse 99 D-5650 SOLINGEN 1
ENNING Klaus-Dieter ol.-Phys.	GKSS Forschungszentrum Geesthacht GmbH Max-Planck-Strasse Postfach 1160 D-2054 GEESTHACHT
ERRMANN Dieter Dr.	Beratungsstelle für Umweltfragen mbH Bruno-Wille-Strasse 56 D-O-1162 BERLIN
L Peter Dr. rer.nat.	Forschungszentrum Jülich GmbH ASS Postfach 1913 D-5170 JÜLICH
NGMANN Hans Dipl. Ing.	Fachhochschule GIESSEN-FRIEDBERG Wiesenstrasse 14 D-6300 GIESSEN
OFFMANN Christoph	Münchner Apparatebau Mehlbeerstrasse 2 D-8028 TAUFKIRCHEN 1

GERMANY

HOLL Matthias Dipl. Phys.	RWE Energie Aktiengesellschaft Kraftwerk Mühlheim-Kärlich Am guten Mann Postfach 1432 D-5403 MÜHLHEIM-KÄRLICH
HOLZHEIMER Claus	Fa. Hörmann Entwicklungs GmbH Hauptstrasse 45-47 D-8011 KIRCHSEEON
HORLBECK B. Dipl. Phys.	DBE Woltorfer Strasse 74 D-3150 PEINE
HÜBEL Klaus RD	Bayerische Landesanstalt f. Wasserforschung Kaulbachstr. 37 D-8000 MÜNCHEN 22
HÜTHER Fritz-Joachim Dr.	Universität Marburg Fachbereich 14, Kernchemie Hans-Meerwein-Strasse D-3550 MARBURG
ILMSTAEDTER Volker Dr. Ing.	Hessische Landesanstalt für Umwelt Kasinostrasse 60 D-6100 DARMSTADT
IWATSCHENKO Michael Dr. FAG	Kugelfischer Georg Schäfer KGaA Fraueauracher Str. 98 D-8520 ERLANGEN
JOCKWER Norbert Dr.	GSF-Forschungszentrum Institut für Tieflagerung Theodor-Heuss-Str. 4 D-3300 BRAUNSCHWEIG
KADEN Michael Dipl.-Phys.	Kraftwerks-u. Anlagenbau AG Berlin Bruno-Dietze-Ring 51 D-O-8355 NEUSTADT
KARDELS Dieter	Industrieanlagen-Betriebsgesellschaft Einsteinstrasse 2 D-8012 OTTOBRUNN
KASPAR Dieter GD	Ministerium f. Umwelt Baden-Württemberg Kernerplatz Postfach 10 34 39 D-7000 STUTTGART 1
KASPARI Werner Dipl. Phys.	Kraftwerks- und Anlagenbau AG Allee der Kosmonauten 32 D-O-1140 BERLIN
KIRCHNER Gerald Dr.	Universität BREMEN Fachbereich 1 Postfach 33 04 40 D-2800 BREMEN 33
KIRSCH Sabine	Hochschule für Technik und Wirtschaft des Saarlandes Göbenstrasse 40 D-6600 SAARBRÜCKEN
KIRTZEL Hans-Jürgen	METEK GmbH Lederstrasse 28 D-2000 HAMBURG 54
KNOBUS Bernd Dr.	Sächsisches Landesamt für Umwelt und Geologie Rudolf-Renner-Platz 14 D-O-8028 DRESDEN
KÖHLER Gerhard Dipl. Phys.	Kraftanlagen AG im Breitspiel 7 Postfach 10 34 20 D-6900 HEIDLEBERG 1
KRAUSE Walter J. Dr.rer.nat.	Bundesanstalt für Gewässerkunde Kaiserin Augusta Anlagen 15-17 Postfach 309 D-5400 KOBLENZ

GERMANY

RUSE Jochen	SINA Industrieservice GmbH Bayernstrasse 53 D-7530 PFORZHEIM
ÜNKEL Renate Dr.	Gesellschaft für Schwerionenforschung Planckstrasse 1 D-6100 DARMSTADT
NGGUTH Karl-Günther Dr.	Kernforschungszentrum Karlsruhe Hauptabt. Sicherheit / Umweltschutz Postfach 36 40 D-7500 KARLSRUHE
UER Raimund Dr. rer.nat.	Umweltbehörde HAMBURG Umweltuntersuchungen Gazellenkämp 38 D-2000 HAMBURG 34
HMANN Karl-Heinz	TÜV-Bayern-Sachsen e.V. G2 ETS 50 Westendstrasse 199 D-8000 MÜNCHEN
NHARDT Joachim Dipl. Ing.	Bayrisches Staatsministerium für Landesentwicklung und Umweltfragen Rosenkavalierplatz 2 D-8000 MÜNCHEN
ER Fritz Ing. (grad.)	Bundesminister für Umwelt und Reaktorsicherheit, RS II 5 Postfach 12 06 29 D-5300 BONN
AJEWSKI Manfred	STI-Vertriebs-GmbH Viktoriastrasse 5 D-5632 WERMELSKIRCHEN 1
ARTENS Klaus Dr.	Ges.f. Anlagen- und Reaktorsicherheit Schwertnergasse 1 Postfach 10 16 50 D-5000 KÖLN 1
ASSMEYER Reinhard Dr.	Ges. f. Anlagen- und Reaktorsicherheit Schwertnergasse 1 Postfach 10 16 50 D-5000 KÖLN 1
AUSHART Bärbl	Lab. Prof. Dr. Berthold GmbH & Co. Calmbacherstrasse 22 Postfach 10 01 63 D-7547 WILDBAD 1
AUSHART Rupprecht Dr.	Labor Prof. Dr. Berthold GmbH & Co. Calmbacherstrasse 22 Postfach 10 01 63 D-7547 WILDBAD 1
AYER Ulrich Dipl. Ing.	SIEMENS AG Abt. KWU-UB-F512 Hammerbacherstr. D-8520 ERLANGEN
ETZGER Lothar . Dipl. Phys.	Hessisches Ministerium für Umwelt, Energie u. Bundesangelegenheiten Mainzer Strasse 80 D-6200 WIESBADEN
ETZKE Roland Dipl. Biol.	TÜV Bayern-Sachsen e.V. Westenstrasse 199 D-8000 MÜNCHEN
EYER Herbert Dipl. Ing.	GSF-Forschungszentrum Umwelt u. Gesundheit GmbH Institut für Tieflagerung Schachtanlage Asse D-3346 REMLINGEN

GERMANY

PULZER Roland Dipl. Phys.	Kraftwerks- und Anlagenbau AG Allee der Kosmonauten 32 D-O-1140 BERLIN
RALL Anna Maria Dipl. Met.	TÜV Bayern-Sachsen e.V. G2-ETS 50 Westendstr. 199 D-8000 MÜNCHEN 21
RASP Winfried Dipl. Ing.	Physikalisch-Technische Bundesanstalt Lab. 6.23 Bundesallee 100 D-3300 BRAUNSCHWEIG
REICHERT Alexander	Kernforschungszentrum Karlsruhe Hauptabteilung Sich Strahlenschutz Postfach 3640 D-7500 KARLSRUHE 1
REICHERT Benno	Herfurth GmbH Beerenweg 6-8 D-2000 HAMBURG 50
REINHOLZ Uwe Dipl. phys.	Bundesanstalt für Materialforschung und -Prüfung Unter den Eichen 87 D-1000 BERLIN 45
RIECK Wolfgang Dr.	FAG-Kugelfischer Georg Schäfer KGaA Strahlenmesstechnik Frauenauracher Strasse 98 D-8520 ERLANGEN
RÖBIG Gerhard Dr.	Siemens AG Bereich Energieerzeugung (KWU) Abt. R 262 Postfach 10 10 63 D-6050 OFFENBACH
RÖSLER Holger Dr.	Hamburgische Electricitäts-Werke AG Überseering 12 D-2000 HAMBURG 60
ROSENSTOCK Wolfgang Dr.	Fraunhofer-Institut INT Postfach 1491 D-5350 EUSKIRCHEN
RUDOLF Arthur	Dornier GmbH Abt. VF1 432 Postfach 1420 D-7990 FRIEDRICHSHAFEN 1
RÜHLE Horst Dipl. phys.	Inst. f. Wasser, Boden- und Lufthygiene des Bundesgesundheitsamtes Postfach 330013 Corrensplz D-1000 BERLIN 33
SACKMANN Siegfried Dr.rer.nat.	Bundesamt für Strahlenschutz KTA-Geschäftsstelle Albert-Schweitzer-Strasse 18 D-3320 SALZGITTER 1
SALFELD Hans-Christoph Dipl.Phys.	Niedersächsisches Landesamt für Immissionsschutz Göttinger Str. 14 D-3000 HANNOVER 91
SCHIFFMANN Edgar Dipl.Ing.	Staatliches Materialprüfungsamt Nordrhein-Westfalen Marsbruchstrasse 186 D-4600 DORTMUND 41
SCHINK Franz Dipl. Ing.	Brennelement Zwischenlager Ahaus Ammeln 59 D-4422 AHAUS
SCHMITT Thomas	Hochschule für Technik und Wirtschaft der Saarlandes Göbenstrasse 40 D-6600 SAARBRÜCKEN

GERMANY

DREKER Harm-Wulf -Ing.	Kraftanlagen AG Im Breitspiel 7 Postfach 10 34 20 D-6900 HEIDELBERG 1
LLMANN-COERS M.	Forschungszentrum Jülich Abt. ASS-UW Postfach 1913 D-5770 JÜLICH
LLER Jürgen Dr.	Ministerium f. Arbeit, Soziales, Jugend, Gesundheit und Energie Schlesw.-Holst. Brunswikerstr. 16-22 Postfach 1121 D-2300 KIEL 1
LLER-GLEWE Jürgen er.nat.	TÜV-Hannover Leiter Zentralabt. Strahlenschutz Am TUeV 1 D-3000 HANNOVER-DÖHREN
LLER-KIEMES Bernd	Hessisches Ministerium für Umwelt Albertusstrasse 15 D-6530 BINGEN
LER-LYDA Ingo Dr.	GSF-Forschungszentrum GmbH Institut für Tieflagerung Theodor-Heuss-Strasse 4 D-3300 BRAUNSCHWEIG
NDIGL Stefan Dr.	Bundesamt für Strahlenschutz Institut für Strahlenhygiene Ingoldstädter Landstr. 1 D-8042 NEUHERBERG / München
Alfred Dipl. phys.	Landesanstalt für Umweltschutz Baden- Württemberg Hertzstrasse 173 D-7500 KARLSRUHE 1
JBURGER Erich -Phys.	ADMINTEC GmbH Bayernstrasse 53 D-7530 PFORZHEIM
KERMÜLLER Ralf	Canberra-Packard GmbH Hahnstrasse 70 D-6000 FRANKFURT AM MAIN
PADOPOULOS Demetre	Kernforschungszentrum Karlsruhe Hauptabteilung Sicherheit / Umweltschutz Postfach 36 40 D-7500 KARLSRUHE
VLYTSCH Stephan -Ing.	Admintec GmbH Niederlassung Nord Abt. JSS Bayernstrasse 53 D-7530 PFORZHEIM
GEL Klaus Dr. Dipl.Phys.	Universität-GH SIEGEN Fachbereich 7 (Physik) A.-Reichweinstr. Postfach 101240 D-5900 SIEGEN
TH Dietger Dipl. Ing.	Hessische Landesanstalt für Umwelt Ludwig-Mond-Strasse 33 D-3500 KASSEL
OKERT Konrad Dr.rer.nat.	Technische Universität Dresden Institut für Strahlenschutzphysik Mommsenstr. 13 D-O-8027 DRESDEN
RROS Nikolaos er.nat.	Philipps-Universität Marburg Fachbereich 14 Kernchemie Hans-Meerwein-Strasse D-3550 MARBURG

GERMANY

SCHOLZ Herwig Dr. Akad. Direktor	Universität Oldenburg Ammersländer Heerstrasse 114-118 Postfach 2503 D-2900 OLDENBU
SCHREYER Hanns Dipl.Phys.	TÜV Südwestdeutschland Gottlieb-Daimler-Strasse 7 D-7024 FILDERSTADT
SCHÜTZ	Herfurth GmbH Beerenweg 6-8 D-2000 HAMBURG 50
SCHULZE Hartmut Dipl. Phys.	Deutsche Gesellsch. zum Bau und Betrieb von Endlagern Endlager Morsleben D-O-3241 MORSLEBE
SCHWANZER Sylvia Dipl. Phys.	Wiederaufarbeitungsanlage Karlsruhe Betriebsgesellschaft mbH Postfach 1263 D-7514 EGGENSTEIN-LEOPOLDSHAFEN 2
SCHWARZ Herwig	Bundesamt f. Wehrtechnik u. Beschaffung Postfach 73 D-5400 KOBLENZ
SEIDER Werner	Kernkraftwerk Philippsburg Postfach 1140 D-7522 PHILIPPSBURG
SEYFFER Ulrich Dipl. Ing.	(FH) Gemeinschaftskernkraftwerk Neckar Postfach D-7129 NECKARWESTHEIM
SÖLTER Barbara Dipl.Biochem.	Deutsche Gesellschaft für zerstörungsfreie Prüfung e.V. Unter den Eichen 87 D-1000 BERLIN 45
STADGE Godehard	Zentralstelle für Sicherheit des Landes Nordrhein- Westphalen Ulenbergstrasse 127-131 D-4000 DÜSSELDORF
STEINKOPFF Th. Dr.	Deutscher Wetterdienst Keiserleistrasse 42 D-6050 OFFENBACH am Main
TILLACK Susanne Dipl. Phys.	Umweltministerium des Landes Brandenburg Einsteinstrasse 42-46 D-O-1561 POTSDAM
TOTTEWITZ Karlheinz	Erzgebirge Wasser/Abwasser AG Theresenstrasse 13 D-O-9010 CHEMNITZ
TRÄBER Karl-Heinz Dipl.-Ing.	Vacutec GmbH Dornblüthstrasse 14 D-O-8021 DRESDEN
TSCHAUDER Wilhelm Wiss. Dir.	Bundesamt für Zivilschutz (BZS) Deutschherrenstrasse 93-95 D-5300 BONN 2
TSCHERNER Manfred Dipl. phys.	TÜV Rheinland e.V. Am Grauen Stein / Konst.Wille-Str. 1 Postfach 10 17 50 D-5000 KÖLN 1
VÖLZ Eckhard Dr.	Preussen-Elektra Treschkowstrasse 5 D-3000 HANNOVER 91

GERMANY

EBER Max Dr.	Johannes-Gutenberg-Universität Mainz Institut für Kernchemie F. Strassmann-Weg 2 Postfach 3980 D-6500 MAINZ
ERSHOFEN Herbert rer.nat.	Physikalisch-Technische-Bundesanstalt Postfach 33 45 D-3300 BRAUNSCHWEIG
CKE Andreas Dr.	Kernforschungszentrum Karlsruhe Hauptabteilung Sicherheit Postfach 36 40 D-7500 KARLSRUHE 1
OLF Frank	WISMUT GmbH Jagdschänkestrasse 29 D-O-9030 CHEMNITZ
OLF Horst Dipl.Ing.	Ministerium f. Wirtschaft, Mittelstand und Technologie Nordrhein-Westphalen Haroldstr. 4 D-4000 DÜSSELDORF 1
UNSCH Klaus-Dieter Dr.	TÜV Bayern-Sachsen Abt G2 ETS 50 Westendstrasse 199 D-8000 MÜNCHEN
PPE Dietmar Dr.	Bundesamt f. Strahlenschutz (Aussenstelle Berlin) Postfach 10 0149 D-3320 SALZGITTER
HNER Herbert Dr.	PreussenElektra AG Treschkow-Strasse 5 D-3000 HANNOVER
UBECK Norbert Dr.	Höchst AG Radiochemisches Laboratorium Postfach 80 03 20 D-6230 FRANKFURT am Main 80

FRANCE

ARNAUD Michel	CEA-COGEMA Etablissement de LA HAGUE BP 52 F-50444 BAUMONT-La HAGUE Cédex
BALLEREAU Pierre	107, Avenue de Paris F-36000 CHATEAUROUX
BERTEL André	CEA / Dir. des applications militaires Centre d'Etude de Ripault BP 16 F-37260 MONTS
BETIS Jacques	COGEMA La HAGUE Service Prévention Radioprotectic BP 52 F-50444 BEAUMONT-HAGUE Cédex
BIRRAUX Claude	DEPUTE de Haute Savoie 5, Avenue de la République F-74100 ANNEMASSE
BLAISE Philippe	CEA / COGEMA La Hague BP 52 F-50444 BEAUMONT-LA HAGUE Cédex
BLANC	Service Central de Protection contre les Rayonnements Ionisants BP 35 F-78110 LE VESINET
BOSSARD Jean-Luc	Electricité de France 13-27, Esplanade Charles de Gaul F-92057 PARIS la DEFENSE
BOUSQUET-CLAYEUX Geneviève	CEA / Institut de Protection et de Sureté Nucléaire BP 6 F-92265 FONTENAY-AUX-ROSES Cédex
BUECHNER Werner	CEA / Centre d'Etude Nucléaire de Fontenay-aux-Roses BP 6 F-92265 FONTENAY-AUX-ROSES Cédex
CARRE M.	Centrale nucléaire de FESSENHEIM BP 15 F-68740 FESSENHEIM
CHASSARD-BOUCHAUD Colette	Université Pierre et Marie Curie Laboratoire de Biologie Physiologie 4, Place Jussien BP 006 F-75252 PARIS Cédex 05
CLECH Albert	CEA / COGEMA Etablissement de MARCOULE BP 170 F-30206 BAGNOLS-SUR-CEZE Cédex
COPPE Philippe	EDF Dir. de l'équipement - Service études et projets thermiques et nucléaires 12-14 Av. Dutrievoz F-69628 VILLEURBANNE Cédex
CORTELLA J.	CEA / Centre d'études de VALDUC F-21120 IS SUR TI
COUCHET Jacques	COGEMA Pierrelatte BP 16 F-26701 PIERRELATTE
COULON René	CEA / Institut de Protection et de Sureté Nucléaire / DPH CERP BP 6 F-92265 FONTENAY-AUX-ROSES Cédex
CRABOL Bernard	CEA/ Institut de Protection et de Sureté Nucléaire/ DPEI F-92265 FONTENAY-AUX-ROSES Cédex

FRANCE

.QUIER Luc	CEA / Institut de Protection et de Sureté Nucléaire / DPEI / SERE BP 1 F-13108 ST-PAUL-LEZ-DURANCE Cédex
NIER Arlette	22, Rue Maurice Labrousse F-92160 ANTONY
GONNE Pierre-Henri	48, rue Pernety F-75014 PARIS
MAIN P.	Institut de Protection et de Sureté Nucléaire / CEA - Département de Protection de l'Environnement et des installations BP F-50444 BEAUMONT HAGUE Cédex
T Paul	CEA / Institut de Protection et de Sureté Nucléaire BP 6 F-92265 FONTENAY-AUX-ROSES Cédex
MANN Philippe	Electricité de France DEPT/DSRE 6 Rue Ampère BP 114 F-93203 ST-DENIS Cédex 1
E Jean-Yves	CEA / Centre d'études de VALDUC F-21120 IS-SUR-TILLE
UIN Michèle	Electricité de France Comite de radioprotection 3, Rue de Messine F-75008 PARIS
ENT Henri	SCN Réseau Eurisys Montigny-le-Bretonneux 1 rue des Hérons F-78182 ST-QUENTIN-YVELINES Cédex
ORRE André	Electricité de France DSRE 6, rue Ampère BP 114 F-93203 ST-DENIS Cédex
Françoise	SSP/SPR/CR/CEE COGEMA Etablissement de MARCOULE BP 170 F-30206 BAGNOLS-SUR-CEZE Cédex
ORGUES Ghislaine	Electricité de France Direction de l'Equipement 22, Avenue de Wagram F-75382 PARIS Cédex 08
Michel	CEA Centre d'Etudes de la Vallée du Rhône BP 170 F-30205 BAGNOLS-SUR-CEZE Cédex
ESSE Daniel	CEA / Institut de Protection et de Sureté Nucléaire Département de Protection de l'Environnement et des Installations BP 6 F-92265 FONTENAY-AUX-ROSES Cédex
SE Jean-Charles	SGN Réseau Eurisys Montigny-le-Bretonneux 1 rue des Hérons F-78182 ST-QUENTIN-YVELINES Cédex
SON Olivier	CEA / Institut de Protection et de Sureté Nucléaire/ DPEI / SERE BP F-13108 St-PAUL-LEZ-DURANCE Cédex
BERT Henri	CEA / Institut de Protection et de Sureté Nucléaire / DPEI BP F-13108 ST PAUL-LES-DURANCE Cédex
VIER Jean-Michel	CEA / Institut de Protection et de Sureté Nucléaire / DPEI / SERE BP F-13108 ST PAUL-LEZ-DURANCE Cédex

FRANCE

METIVIER Henri	CEA / Institut de Protection et de Sureté Nucléaire BP 6 F-92265 FONTENAY-AUX-ROSES Cédex
MICHEL Pierre-Marie	COGEMA Etablissement de MARCOULE BP 170 F-30206 BAGNOLS-SUR-CEZE Cédex
MISTRAL Jean-Pierre	COGEMA-Etablissement de Marcoule BP 170 F-30206 BAGNOLS-SUR-CEZE Cédex
PENNEROUX Jacques	CEA/Direction centrale sécurité 31-33, Rue de la Fédérat F-75015 PARIS
PICAT Philippe	CEA / Institut de Protection et de Sureté Nucléaire / DPEI SERE BP F-13108 ST PAUL-LEZ-DURANCE Cédex
PINEIRA Thomas	CEA / Centres d'études nucléaires de Fontenay-Aux-Ros BP 6 F-92265 FONTENAY-AUX-ROSES Cédex
QUINAULT Jean-Marie	CEA / Institut de Protection et de Sureté Nucléaire / DPE SERE BP 1 F-13108 ST-PAUL-LEZ-DURANCES Cédex
RENIER Sylvie	Electricité de France DEPT/EPN 6, rue Ampère BP 114 F-93203 ST-DENIS Cédex 1
ROBE Marie-Christine	CEA / Institut de Protection et de Sureté Nucléaire DPEI / SERAC F-91191 GIF-SUR-YVETTE Cédex
ROBEAU Daniel	CEA/ Institut de Protection et de Sureté Nucléaire / DPEI BP 6 Av.G.Leclerc 60-68 F-92265 FONTENAY-AUX-ROSES Cédex
SCHMIDT Eric	CEA / Institut de Protection et de Sureté Nucléaire BP 6 F-92265 FONTENAY-AUX-ROSES Cédex
SONTAG G.	ARIES 33bis, Bd. Félix Faure F-92320 CHATILLON
TAMAS Marie-Claude	CEA / Institut de Protection et de Sureté Nucléaire Département de Protection de l'Environnement et des Installations BP 6 F-92265 FONTENAY-AUX-ROSES C
THOMASSIN Alain	SGN Réseau EURISYS Montigny-le-Bretonneux 1, rue des Hérons F-78182 ST-QUENTIN-YVELINES Cé
UZZAN Guy	Société Française de Radioprotection BP 72 F-92265 FONTENAY-AUX-ROSES Cédex
VESSERON Philippe	CEA / Institut de Protection et de Sureté Nucléaire BP 6 F-92265 FONTENAY-AUX-ROSES Cédex
WENDUM Denis	EDF - Dir. des Etudes et Recherches Dep. Environnemer 6, Quai Watier F-78400 CHATOU Cédex
ZAPP Robert Directeur	NOVELEC SA BP 147 F-38244 MEYLAN Cédex

OTHERS COUNTRIES

TTO Graeme	Ühlinger-Pfiffner Mini-Instruments Ltd. CMO 8RN England BURNHAM on Crouch, Essex
CORT M. Dr.	CEC Joint Research Center JRC-ISPRA I-21020 ISPRA / ITALIA
DD David	National Radiological Protection Board OX11 ORQ DIDCOT Oxfordsh. Chilton U.K.
VRILOV Sergei	Nuclear Safety Institut (IBRAE) Academy of Sciences Leningradskoe Shosse 116 App. 105 MOSCOU
EMA T.	Ühlinger-Pfiffner Veenstra Instruments BV Madame-Curiewei 1 NL-8500 AC JOURE
NEVSKY Mikhail F. Dr.	Institute for Nuclear Safety Russian Academy of Sciences 52, B. Tulskaya 113191 MOSCOW, RUSSIA
ELEV Vladimir P. Dr.	Institute for Nuclear Safety Russian Academy of Sciences 52, B. Tulskaya 113191 MOSCOW, RUSSIA
RO RODRIGUEZ Conrado Dr.	Université De Extramadura Dept. of Physics, Facultad de Veterinaria Ctra. de Trujillo s/n E-10071 CACERES (Espagne)
LLER Hansjörg Prof.	Reaktorinstitut Universitaet Graz Steyerg. 17 A-8010 GRAZ
VARRO Elisa Dr.	IFIC Universitad de Valencia Avda. Dr. Moliner 50 E-46100 BURJASOT / Espagne
CHINGER Ulrike Dr.	Zentralanstalt für Meteorologie und Geodynamik Hohe Warte 38 A-1190 WIEN
GAL Michael Dr.	Ministry of Agriculture,Fisheries & Food Ergon House - c/o Nobel House 17 Smith Square SW1P 3JR LONDON U.K.
CHURLOVITS Manfred l. Dr.	Atominstitut Der Österr. Universitäten Schüttelstrasse 115 A-1020 WIEN
Z CARREIRO Carolina Dr.	LNETI Dep.Protecao e Seguranca Radiologica Estrada Nacional 10 P-2685 SACAVEM (Portugal)

Alphabetic Index of Authors

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